VOLUME III

TRANSCRIPT OF RECORD

Supreme Court of the United States

OCTOBER TERM, 1960

No. 315

POWER REACTOR DEVELOPMENT COMPANY, PETITIONER,

vs.

INTERNATIONAL UNION OF ELECTRICAL, RADIO AND MACHINE WORKERS, AFL-CIO, ET AL.

No. 454

UNITED STATES, ET AL., PETITIONERS,

vs.

INTERNATIONAL UNION OF ELECTRICAL, RADIO AND MACHINE WORKERS, AFL-CIO, ET AL.

ON WRITS OF CERTIORARI TO THE UNITED STATES COURT OF APPEALS FOR THE DISTRICT OF COLUMBIA CIRCUIT

NO. 315 PETITION FOR CERTIORARI FILED AUGUST 12, 1960

NO. 454 PETITION FOR CERTIORARI FILED SEPTEMBER 29, 1960

CERTIORARI GRANTED NOVEMBER 14, 1960

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[fol. 3920] In evidence January 8, 1957, Tr. 42

BEFORE THE ATOMIC ENERGY COMMISSION

Docket No. F-16

In the Matter of

POWER REACTOR DEVELOPMENT COMPANY

TESTIMONY OF ERNEST R. ACKER, VICE-PRESIDENT, POWER REACTOR DEVELOPMENT COMPANY

Background and Qualifications

My name is Ernest R. Acker. I am a member of the Board of Trustees of Power Reactor Development Company (PRDC). I am also a Vice-President of PRDC, a member of the Executive Committee and Chairman of the Financial Committee. I am President of Central Hudson Gas & Electric Corporation, which is a member of PRDC.

I graduated from Cornell University with a degree in mechanical engineering in 1917. During World War I, I served in the Chemical Warfare Service with the rank of First Lieutenant. In 1919 I commenced my employment with Central Hudson Gas & Electric Company, predecessor of the present corporation, and I have been with this organization continually since that date. I began my employment in the Engineering Department and subsequently became General Superintendent of the Kingston District of the Company and later General Commercial Manager of the Company. I have been a Director of Central Hudson Gas & Electric Corporation since 1927 and have held the [fol. 3921] position of President and General Manager of that corporation since 1932. I am a member of Tau Beta Pi, an honorary engineering society. I have been President of Edison Electric Institute and the American Gas Association. I am also a member of the Board of Directors of Atomic Power Development Associates, Inc. (APDA), and have been since its organization.

Organization and Background of Power Reactor Development Company

PRDC is a Michigan non-profit membership corporation organized to study and demonstrate the economic and technical feasibility of producing electricity from nuclear fuels through the design, construction and operation of a developmental nuclear power reactor. PRDC is now engaged in the design, development and construction of a full scale fast neutron breeder reactor at a site on Lake Erie in Monroe County, Michigan. This reactor will provide steam which will be used for the production of electricity in facilities owned by The Detroit Edison Company. The entire plant will be known as the Enrico Fermi Atomic Power Plant. Copies of the Articles of Incorporation and By-Laws of PRDC are included as Exhibit III and Amended Exhibit IV to the Application, as amended, for a license under Section 104 b of the Atomic Energy Act of 1954, filed by PRDC. PRDC is a membership corporation and as such cannot issue shares of stock or any other evidence of ownership. The members of PRDC are twenty-one in number and are all corporations.

[fol. 3922] There are thirteen operating utility company members as follows:

Central Hudson Gas & Electric Corporation
Cincinnati Gas & Electric Company
The Columbus and Southern Ohio Electric Company
Consumers Power Company
Delaware Power & Light Company
The Detroit Edison Company
Iowa-Illinois Gas & Electric Company
Long Island Lighting Company
Philadelphia Electric Company
Potomac Electric Power Company
Rochester Gas & Electric Corporation
The Toledo Edison Company
Wisconsin Electric Power Company

There are seven industrial company members as follows:

Allis-Chalmers Manufacturing Company
The Babcock & Wilcox Company

Burroughs Corporation
Combustion Engineering, Incorporated
Fruehauf Trailer Company
Holley Carburetor Company
Westinghouse Electric Corporation

Another member is ~~a service~~ company, Southern Services, Inc. representing Alabama Power Company, Georgia Power Company, Gulf Power Company, Mississippi Power Company, each of which is an operating utility and a contributor to PRDC, and the parent of those five companies, The Southern Company. Southern Services, Inc., also provides the Trustee representative.

The purpose of PRDC is to do research and development in the field of the peacetime application of atomic energy for the production of electric power through the construction and operation of a developmental power reactor, and to make information acquired from these research and de- [fol. 3923] velopment activities available, not only to its members but also to the Atomic Energy Commission (AEC) for the use of the public, and to all others who may desire such information, subject solely to AEC security regulations. PRDC is not organized for profit. The Articles of Incorporation provide that no part of the net earnings of PRDC may inure to the benefit of any member, any private individual, or any corporation. They provide further that members may receive nothing on dissolution, since in the case of dissolution any remaining assets must be turned over to an organization or organizations operated exclusively for educational or scientific purposes. Under the By-Laws of PRDC members are prohibited from obtaining any special privilege to use patents which may be developed by PRDC, and the By-Laws also require that all such patents be made freely available to anyone who may properly request their use.

Memberships in PRDC are non-transferable, and changes can occur only when new members are elected or old members resign, and members may be elected to membership and resign membership. There is no written instrument evidencing membership. Each member has one vote regard-

less of the amount of its financial contribution to the activities of PRDC.

The affairs of PRDC are conducted by a Board of Trustees, twenty-seven in number, who are elected by the members. No member may be represented on the Board of Trustees by more than one person. The Trustees are as follows:

[fol. 3924] Ernest R. Acker, President
Central Hudson Gas & Electric Corporation

Dr. Robert F. Bacher, Director
Norman Bridge Laboratory
California Institute of Technology

Walter C. Beckjord, President
The Cincinnati Gas & Electric Company

George Bisset, Vice-President
Potomac Electric Power Company

Walker L. Cisler, President
The Detroit Edison Company

John S. Coleman, President
Burroughs Corporation

Stuart Cooper, Chairman
Delaware Power & Light Company

Errol W. Doebler, President
Long Island Lighting Company

George D. Ellis, Vice-President
Combustion Engineering, Incorporated

Robert E. Ginna, President
Rochester Gas & Electric Corporation

Dr. Lawrence R. Hafstad, Vice-President in Charge of Research
General Motors Corporation

Daniel C. Hickson, Vice-President
Bankers Trust Company

George M. Holley, Jr., President
Holley Carburetor Company

Alfred Iddles, President
The Babcock & Wilcox Company

Charles E. Ide, President
The Toledo Edison Company

[fol. 3925] Dan E. Karn, President
Consumers Power Company

Edward L. Love
Chase Manhattan Bank

Jackson Martindell, President
American Institute of Management

Harry M. Miller, President
Columbus & Southern Ohio Electric Company

Alexander C. Monteith, Vice-President
Westinghouse Electric Corporation

R. George Rincliffe, President
Philadelphia Electric Company

J. Frank Roberts, Vice-President
Allis-Chalmers Manufacturing Company

Herbert J. Scholz, President
Southern Services, Inc.

Dean E. Blythe Stason
Law School, University of Michigan

Gould W. VanDerzee, President
Wisconsin Electric Power Company

Charles H. Whitmore, President
Iowa-Illinois Gas & Electric Company

Dr. Theodore P. Wright, Vice-President
for Research
Cornell University

It will be observed that each member company, except Fruehauf Trailer Company, is represented by a trustee. Each trustee is an officer of the company he represents, and in addition there are trustees who are unconnected with any member and who represent the fields of education, finance and scientific research.

[fol. 3926] The Board of Trustees elects the principal officers of the company. They are as follows:

Walker L. Cisler, President
(The Detroit Edison Company)

R. George Rincliffe, Executive Vice President
(Philadelphia Electric Company)

Dan E. Karn, Vice President
(Consumers Power Company)

Herbert J. Scholz, Vice President
(Southern Services, Inc.)

Ernest R. Acker, Vice President
(Central Hudson Gas & Electric Corporation)

John A. Lagrou, Treasurer and Assistant Secretary

John F. Anderson, Assistant Treasurer and Assistant Secretary

George E. Olmsted, Secretary and Assistant Treasurer

Errol Doebler
(Long Island Power & Lighting)

The Board of Trustees also functions through the following committees whose respective responsibilities are apparent from their names: The Executive Committee (of which I am a member), The Technical and Engineering Committee, The Administrative Advisory Committee, The Legal Committee, The Financial Committee (of which I am Chairman), and The By-Laws Committee. While one or more trustees serve on each committee, the committees also include others who are not trustees.

The utility company members and contributors provide approximately 17% of the electric generating capacity of [fol. 3927] the investor-owned utility companies in the United States, have a combined earned surplus of approximately $400,000,000, and have combined assets in excess of $2,000,000,000. In addition, the industrial members have very substantial financial resources and are important representatives of various segments of American industry. It is the purpose of the members of PRDC to construct and operate the reactor portion of the Enrico Fermi Atomic Power Plant in order to provide themselves and the country with the knowledge that will be gained in this important area of the utilization of nuclear materials.

The origin of the PRDC fast breeder reactor goes back to April, 1951, when The Detroit Edison Company and Dow Chemical Company entered into a study agreement with the AEC to conduct research, looking to the feasibility of generating electricity by the use of fissionable materials.

In the period between 1952 and 1955 certain other companies joined the study group initiated by The Detroit Edison Company and Dow Chemical Company. In 1954 Dow Chemical Company withdrew from the group to concentrate on the application of atomic energy to chemical processes. APDA was organized as a New York membership corporation in March of 1955 to carry on the work which had previously been conducted by this study group. As a result of its studies, APDA determined that the fast neutron breeder reactor held great promise because of conservation of nuclear fuels possible through the use of this type of reactor. As Dr. Bethe has stated in his testimony, the [fol. 3928] development of breeder reactors is of the utmost importance to an economic use of nuclear fuels for peacetime purposes because of the fuel conservation possible.

The adoption by Congress of the Atomic Energy Act of 1954 to encourage participation of private industry in advancing the peacetime application of the atom made possible an extended industrial effort in this area. In January, 1955, the AEC announced its Power Demonstration Reactor Program and invited industry to submit proposals for the construction of reactors of various types. On March 30, 1955, The Detroit Edison Company, on behalf of a group of companies, all of which are now members of PRDC, submitted to the AEC a proposal under this program to design, construct, own and operate a developmental fast neutron breeder reactor. On August 8, 1955, the AEC announced that the March 30 proposal was acceptable as a basis for further negotiation. Copies of the proposal made by The Detroit Edison Company to the AEC and of the acceptance of the proposal as a basis for negotiation are contained in the PRDC Application for License as Exhibits I and II, respectively. PRDC and the AEC are now negotiating an operating agreement under the Power Demonstration Reactor Program.

I think I should make clear the fact that Atomic Power Development Associates, Inc. (APDA) and Power Reactor Development Company (PRDC) are completely separate and distinct companies. I have already described PRDC [fol. 3929] in some detail. APDA is primarily a technical and design research organization in the field of fast breeder

reactors, and is responsible for the conceptual design of the PRDC reactor. Many of the twenty-one member companies of PRDC are also members of APDA, although not all of them are, and APDA has many members which are not members of PRDC.

APDA, working with PRDC on the fast breeder developmental reactor, is performing design, technical and research functions. APDA will also conduct non-nuclear tests of certain reactor components, and upon the conclusion of such tests will transfer certain of the tested equipment and devices to PRDC for incorporation in its reactor. Arrangements between PRDC and APDA have been forma[illegible]d in a contract dated December 14, 19[illegible], included as Exhibit XXI to the PRDC Application for Lic[illegible]nse and incorporated in this testimony by reference.

The management of PRDC has been guided and will continue to be guided by the design, technical and safety recommendations received from APDA from time to time. PRDC will also conduct the pre-operational and other tests recommended by APDA, in consultation with PRDC consultants, and will adopt start-up, operational and other procedures in accordance with such recommendations. Much of the testimony to be given in this proceeding with respect to the research and test program for the Enrico Fermi reactor is accordingly presented by two APDA employees who are particularly familiar with this area of APDA work, Mr. A. Amorosi and Mr. W. J. McCarthy.

[fol. 3930] On January 7, 1956, PRDC filed an Application for License under Section 104 b of the Atomic Energy Act of 1954 to construct and operate a fast neutron breeder reactor. This Application has been amended from time to time and is now pending. On August 4, 1956, the AEC granted PRDC a provisional construction permit.

PRDC Safety Policy

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Public safety, therefore, has been and is of prime concern to the PRDC management. We are completely satisfied that there is no public risk whatever involved in going forward with the plant construction work at this time. In addition, we will certainly not place the plant in operation [fol. 3931] unless the safety experiments to be conducted

satisfy both the AEC and the staff and consultants of PRDC and APDA that the reactor can be operated without endangering the public health and safety.

Furthermore, the management of PRDC believes that any financial risk in proceeding with the immediate construction of the reactor—and no other risk is involved—is fully justified by the time savings involved. By proceeding with both the construction and the research and experimental program at the same time, we should be able to save several years in the time required to place in operation a demonstration power reactor of significant size which we believe will point the way to the economically feasible production of electric power through the use of nuclear materials. We believe that this time saving is important to the advancement of the technology of nuclear power production, as well as to the preservation of American leadership in the peaceful application of atomic energy.

* * * * * * *

[fol. 3947]

Management Responsibilities. As has been noted heretofore, the management of PRDC did not authorize the filing of the Application for License with the AEC until it was assured by its staff and consulting technicians, engineers and physicists that the health and safety of the public could not possibly be endangered by the construction of the fast breeder reactor at its proposed site in Monroe County, Michigan, and that any problems requiring solution with respect to the operation of the reactor could be satisfactorily resolved. The management of PRDC is, of course, aware that further important research and experiments are to be conducted during the construction period, and in fact has arranged for the conduct of a substantial amount of such research and experimental work, in order to provide further experimental evidence that its reactor can, when constructed, be operated safely. We consider that the most
[fol. 3948] important contribution we can make in expediting the development of the use of atomic energy for the production of ~~electricity~~ is to proceed with the construction of the proposed fast breeder reactor in reliance upon the advice of our staff and consultants. Before the plant is placed in operation all essential safety experiments will have been conducted. We will not place the plant in opera-

tion unless these experiments satisfy both the AEC and the staffs and consultants of PRDC and APDA that the reactor can be operated without endangering the health and safety of the public. We are confident that this will be the case and that it is in the best interest of the public for the construction of this project to go forward without unnecessary delay.

[fol. 3950] In evidence January 8, 1957, Tr. 43

BEFORE THE ATOMIC ENERGY COMMISSION

Docket No. F-16

In The Matter Of

POWER REACTOR DEVELOPMENT COMPANY

TESTIMONY OF HANS A. BETHE, PROFESSOR OF PHYSICS, CORNELL UNIVERSITY

(1) *Biography and Qualifications*

My name is Hans A. Bethe. I am John Wendell Anderson Professor of Physics at Cornell University. I was born in Strassburg, Alsace-Lorraine, on July 2, 1906. I studied at the Universities of Frankfort and Munich and obtained the degree of Doctor of Philosophy in Physics from the University of Munich in 1928. From 1928 through 1933 I taught theoretical physics at the Universities of Frankfort, Stuttgart, Munich and Tubingen. From 1933 through 1935 I was lecturer at the Universities of Manchester and Bristol in England. I came to Cornell University as an assistant professor in 1935 and have been a full professor since 1937.

My general field is theoretical physics. My work from 1934 to 1940, and again since 1946, has been chiefly in nuclear physics. In 1934 intensive study of this subject was just beginning. Many experimental data were constantly obtained, and one of my main activities was to correlate these data, to interpret them and to obtain from them a theoretical understanding of the structure of atomic nuclei

and of the reactions between them. In 1936 and 1937 I wrote three comprehensive review articles, together of the length of an average book, on nuclear physics. These articles were frequently reprinted and served for about 15 years as the standard textbook from which students as well as active research workers in the field obtained their information, in this country as well as abroad.

[fol. 3951] A large fraction of my research work before the War was spent on investigating the properties of neutrons. While not doing any experiments myself, I kept in close contact with the experimental work on neutrons at Cornell University. Our University was a pioneer in this field; in particular we developed the first accurate neutron velocity selector to study the resonances in the interaction of slow neutrons with nuclei.

I also worked on the energy generation in stars by means of nuclear reactions. My theory of this process, first developed in 1938, is generally regarded as the explanation of the large energy production in the sun and other stars. For this work I received two A. Cressy Morrison prizes from the New York Academy of Sciences and the Henry Draper Medal from the National Academy of Sciences.

I am a member, among others, of the National Academy of Sciences, the American Philosophical Society and the American Physical Society. I was President of the latter society in 1954. I hold four honorary degrees of Doctor of Science, from Brooklyn Polytechnic Institute and from the Universities of Denver, Chicago, and Birmingham, England.

Early in the War I became engaged in war research. My first war work, in 1940, was concerned with the penetration of armor by projectiles. This work has been extended by other scientists and the results have been widely used by Ordnance experts, and as far as I know still form the basis of understanding of armor penetration.

Next I was asked to work for Division 8 of the National Defense Research Council which was concerned with shock waves produced by explosives, especially in air and water. During 1941 and 1942, I wrote three papers on the subject of shock waves.

[fol. 3952] Early in 1942 I started work for the Radiation Laboratory at MIT which was concerned with the development of Radar.

In the summer of 1942 I was asked to join a small group of theoretical physicists in Berkeley, California, under the direction of J. R. Oppenheimer which was set up by the "Metallurgical Project" at Chicago. The purpose of the Metallurgical Project was to obtain a chain reaction in uranium and then to use this chain reaction for the manufacture of plutonium. The task of our theoretical group was to study methods by which nuclear explosions could be obtained from fissionable material. This group was the core of the later Los Alamos Scientific Laboratory which was started in April 1943.

I was a member of the Los Alamos Scientific Laboratory from its beginning. When a definite organization of the Laboratory was adopted, I was appointed Chief of the Theoretical Division of the Laboratory and remained in that position until December 1945, after the end of the War. Our Division was responsible for all calculations concerning the atomic bomb, and for most of its conceptual design. In particular we had to devise methods to calculate the diffusion of neutrons in fissionable material and in reflectors, methods which have later proved useful for the study of reactors. We had to calculate the course of a nuclear explosion before any one was ever observed. I am, therefore, thoroughly familiar with calculations of nuclear explosions and with the conditions that are required for such explosions. We also studied ordinary, nonnuclear explosions and the behavior of materials when hit by the shock from ordinary explosions. While all this work was carried out by a large group of people in my division, at one time more than 80 individuals, I kept in close contact with all phases of the work and contributed myself to the solution of many of these problems.

For my work at Los Alamos, I received the Medal of Merit, the highest decoration that can be awarded to a civilian by the United States Government.

[fol. 3953] After the War, I returned to Cornell University. However, I kept in close contact with the development of atomic energy both for war and for peaceful purposes.

In the first place, I became a consultant to the Los Alamos Scientific Laboratory. This involved further work on nuclear weapons, but in addition it served to keep me informed on new methods to estimate nuclear explosions, and on new calculational methods for neutron diffusion. I also kept informed on the newest measurements of nuclear cross sections and participated in the interpretation of some of these which are particularly important for fast breeder reactors.

In addition, I became a consultant to several Laboratories concerned with the design of atomic reactors. I worked particularly intensively with the Knolls Atomic Power Laboratory (KAPL) at Schenectady, from its founding in 1946 until approximately 1950. After this time, KAPL became involved in very special problems relating to Naval Reactors, and although I continued as a consultant I spent much less time on this work. From 1946 to 1950, when I did participate actively, the major project of KAPL was the design of a breeder reactor, which was first intended to work with neutrons of intermediate energy and later with fast neutrons. In this connection, I stimulated and closely followed a program to determine experimentally a certain quantity, the ratio of capture to fission in U^{235} and Pu, which is decisive for the breeding qualities of the breeder reactor. Many novel experimental techniques were invented by the Knolls Atomic Power Laboratory in connection with their fast breeder work. About 1949, I wrote a report for KAPL on the safety of their proposed reactor, in particular on the mild nuclear explosion which could occur as a result of a possible nuclear accident. This report is, to my knowledge, the first report on the theory of nuclear accidents of power reactors.

[fol. 3954] I also consulted for NEPA, a project for nuclear propulsion of aircraft, from 1946 to 1951, and since 1951 for Oak Ridge National Laboratory. In these two contracts, I was chiefly concerned with the theory of the shielding of reactors.

Since the end of the War I have been a member of several committees of the United States Government. They included the Committee on Atomic Energy of the Research and Development Board of the Department of Defense,

from 1952 to 1954, the Nuclear Panel of the Scientific Advisory Board to the U. S. Air Force from 1953 to 1954, and the Scientific Advisory Committee to the Office of Defense Mobilization at present, in addition to minor committees.

I have continued my teaching and my research in pure physics at Cornell University. From September 1955 to July 1956, while on leave from Cornell University, I was Visiting Professor of Physics at the University of Cambridge, England. During this time, I had an opportunity to study the British Atomic Energy effort, in particular on the fast breeder reactor. I was a member of the United States' team to exchange information on the fast reactor with the United Kingdom. I was also a member of the U. S. delegation to the Geneva Conference on Peaceful Application of Atomic Energy in August 1955.

Since 1953 I have been a consultant to Atomic Power Development Associates, Inc. (APDA), and the Study Group which preceded it. This group was first known as the Dow-Detroit Edison Study Group, and then was reorganized as APDA. Pursuant to a contract between APDA and Power Reactor Development Company (PRDC), APDA provides nuclear and other technical services to PRDC. I understand that this contract between APDA and PRDC is included in the PRDC Application for License as Exhibit XXI. In connection with my work for APDA, I have become generally familiar with the proposed design for the Enrico Fermi Reactor, as set forth in APDA-115, included [fol. 3955] as Exhibit XA to the PRDC License Application, as amended. I have worked on some phases of this design as an APDA consultant, and in particular on problems related to its safety and the testing of its safety. In connection with this work, I have become familiar with the work of Argonne National Laboratory on fast reactors, and with the theoretical work of Nuclear Development Associates, Inc. (NDA) in White Plains, New York on the same subject. I have read the testimony to be given in this proceeding by Dr. Norman Hilberry of Argonne National Laboratory, as well as that to be given by Mr. A. Amorosi, Mr. Walter J. McCarthy and Mr. Kenneth Davis.

(2) Principles of Fast Reactors

A reactor is a device in which the energy of the fission of uranium is converted into heat. In a power reactor this heat is removed in such a way as to permit further conversion into electricity by a conventional generator.

The most important ingredient of a reactor is the atomic fuel which is U^{235} in practically all the reactors so far constructed. In the future, plutonium and U^{233} may also be used as reactor fuels. The reactor works by means of neutrons which cause fission in the U^{235} nuclei. The power produced is directly proportional to the rate of this fission. When no neutrons are present in the reactor, no power will be produced.

In each fission of U^{235} about 2.5 neutrons are set free from the nucleus. Some of these neutrons leak out of the reactor, others are captured in the structural material or in the coolant, some will be captured by another nucleus of U^{235}. A certain fraction of the neutrons captured in U^{235} nuclei will cause another fission. If, on the average, one of the neutrons emitted in each fission causes another fission, a self-sustaining chain reaction will result; the number of neutrons will then remain constant, and the power will be steady.

[Tel. 3956] In general, this will not be exactly the case. We define the multiplication factor k which is the number of fissions produced on the average by the neutrons originating from one fission. If $k = 1$, the power is steady. If k is greater than one, the number of fissions in the next generation will be greater than that in the last generation and the power will increase with time. If the multiplication factor k is less than one, the power will decrease with time. The difference $k - 1$ is often called the reactivity or sometimes the excess reactivity of the reactor, and is often denoted by Δk.

The reactivity is controlled ~~by control rods. Three~~ different ways have been proposed and used for controlling a reactor. The first is to vary the amount of fuel in the reactor, i.e. to have some fuel rods which can be moved in and out of the reactor. The second method is to have rods made of a substance which absorbs neutrons strongly, these are called poison rods. Finally, part of the reflector may

be made movable, this then controls the leakage of neutrons from the reactor.

The most common type of control is by poison rods. When such a control rod is inserted into the reactor, more neutrons will be absorbed by it, fewer neutrons will be available to make fission in U^{235} and hence the multiplication factor will drop. The control rod works like a combination of brake and accelerator in an automobile, and just as a car can be accelerated or decelerated at will, so the power of a reactor can be regulated by the control rod. The ease with which the reactors I have seen respond to controls is most impressive.

By the fission of U^{235}, energy is developed. This energy heats the fuel U^{235}. This heat is taken away by the coolant, which is normally a fluid circulating through the reactor. Most of the reactors so far built are cooled with water. Some are cooled with liquid sodium, including the proposed [fol. 3957] PRDC reactor. The sodium will in turn give its heat to water which is thereby transformed into steam, and the steam drives the electric generator. In several designs, including the PRDC design and that of EBR-II, which is to be constructed by Argonne National Laboratory, the transfer of heat from sodium to water is not done directly but through an intermediate heat exchange fluid as an added precaution to prevent release of radioactivity as a result of a chemical reaction between sodium and water.

The proposed PRDC reactor is a fast reactor. This name refers to the speed of the neutrons in the reactor. The neutrons come out of a fission with speeds of about ten thousand miles per second. In some types of reactors they are permitted to collide many times with atomic nuclei, and in each such collision they lose some of their speed, until their speed is reduced to about one mile per second. They are then called thermal neutrons, and reactors in which this process goes on are called thermal reactors. In a thermal reactor most of the fissions are caused by thermal neutrons, and only very few neutrons are able to cause fissions while they are fast. The substance which slows the neutrons down from ten thousand to one mile per second is called the moderator. The best moderators are materials which contain many nuclei of low atomic weight such as hydrogen or carbon, i.e. materials like water or graphite.

In a fast reactor the use of a moderator is avoided as far as possible. Then the neutrons remain fast and will make fissions while their speed is still several thousand miles per second rather than one mile per second. In order to accomplish this all materials containing hydrogen are kept away from the reactor core. This is one of the main reasons why sodium is used as a coolant. Even the amounts of sodium and of structural materials like stainless steel in the reactor core are kept to a minimum, in order not to slow down the neutrons.

[fol. 3958] The purpose of keeping the neutrons fast is to accomplish breeding. This means that the reactor regenerates its own fuel, and if possible produces additional fuel. In order to accomplish this, the materials in the reactor must be so arranged that most of the neutrons which do not cause fission are captured in the common isotope of uranium, U^{238}. If this happens, U^{238} is converted into plutonium which is a very good atomic fuel. This conversion is the process by which plutonium is produced in the big Hanford and Savannah River reactors of the Atomic Energy Commission. However, in these reactors, which are not fast reactors, less plutonium is produced than U^{235} is consumed. It has been shown, both by calculation and by experiment, that in a fast reactor the production of plutonium will be greater than the consumption of U^{235}. Ultimately when plutonium can be used as a fuel for a fast reactor, one hopes to produce in a breeder at least 1.5 plutonium atoms for each plutonium atom used up as fuel. Such a breeder would then be like a coal furnace which makes more coal than it uses up.

It is clear that a breeder will be of the utmost importance for a large scale atomic power economy. It is likely that in 30 to 50 years, a large fraction of our power will be derived from atomic energy. This will only be possible if the supply of atomic fuel is very great. Breeders seem to me indispensable for this purpose. Since power demands increase constantly and at a rapid rate, a very large supply of atomic fuel will be required.

Indeed, one of the most compelling reasons for developing atomic power is to extend our fuel reserves. We hear that our natural oil supply will become insufficient in two or three decades, and that coal reserves, although much

larger, are also limited. Atomic power was discovered just in time to prevent serious worry about our fuel reserves. However, if we use only the isotope U^{235} which is directly fissionable, then all the uranium reserves now known to [fol. 3959] exist in the world will give *less* energy than the known reserves of coal. This is because U^{235} is only .7 per cent of the uranium we find in nature. If we learn to use breeders we can use *all* the uranium instead of just the U^{235}, because a breeder will convert the non-fissionable but fertile material U^{238}, which constitutes 99.3 per cent of natural uranium, into the fissionable material plutonium. This will extend the supply of atomic fuel more than a hundredfold. Only if we succeed in doing this will atomic energy give us a real improvement in our fuel reserve; in this case the uranium reserves will outlast the reserves of our fossil fuels more than 50 times. The development of breeders is, therefore, a most important measure of conservation of our precious atomic fuel.

Other countries have fully recognized the importance of breeders. This is particularly true of Great Britain, which has the greatest need for atomic power because of the high cost of mining coal from partly exhausted coal fields and because of an actual shortage of coal which goes hand in hand with this. Britain has therefore taken the lead in the actual construction of atomic power reactors, and in its program is including fast breeders in an important way. They started their breeder development in 1951, and are now expecting to put a full scale breeder reactor into operation in 1958. This means that they are now several years ahead of our development in this field. Even little Belgium is thinking seriously of the development of breeder reactors. In the United States, we had an initial lead in this field. The Argonne National Laboratory built an experimental breeder reactor of about 1000 kilowatts power as far back as 1950. This reactor, EBR-I, has given much scientific information, and in particular, has definitely established that breeding is possible. But high power breeders will not be completed for several years. As I have already mentioned, Argonne is planning another breeder, EBR-II, this one to be of intermediate power. The only full scale power breeder

on which design has progressed substantially, however, is [fol. 3960] the proposed PRDC reactor.

The construction of this plant, and the simultaneous research and development program of APDA, and of the U. S. Atomic Energy Commission, will help to restore to the United States the leadership in this important section of the atomic power program, and will substantially extend the frontiers of technical knowledge in this area. The simultaneous pursuit of programs of research, development and construction has become standard in the fast-moving field of atomic energy and is necessary in order to keep abreast or ahead of our competitors. The expeditious construction of a full-scale breeder is in my opinion an essential step in any large-scale program of developing economically feasible nuclear power.

(3) *Hazards*

The mere construction of a reactor without the introduction of nuclear fuel involves no hazard to the public, since such construction is performed in the same manner as the construction of any building, industrial plant or power plant. No possibility of the creation of any nuclear hazard exists until nuclear material is present. I can envisage no danger to the United States or its security arising from the construction of a reactor plant.

After nuclear fuel has been introduced into the nuclear reactor and the reactor is put into operation, it is still impossible that the reactor ever explode like an atomic bomb. There is no way in which this could ever happen. The hazards we are speaking of in nuclear reactors do not lie in any violent explosion, but in the possibility that the radioactive fission products which have been formed by the operation of the reactor get released into the atmosphere and thereby cause a radiation hazard for the surrounding area. [fol. 3961] When we talk about these hazards, we must have clearly in mind that we are talking about extremely remote possibilities. The probability of nuclear accidents leading to dangerous release of radioactive substances from a reactor is incomparably smaller than that of the most freakish accidents we occasionally read of in the newspapers and very much smaller than that of the rarest diseases. The

reason why accidents to reactors are so improbable is that they are designed in such a way that one failure, such as carelessness of the operator or failure of one mechanical component of the reactor, by itself can never lead to an accident. Of course, every mechanical component is designed and built with the greatest of care, and the operators are highly trained and competent. But still, an oversight by the operator will not by itself lead to an accident because mechanical devices will take over and insure safety, and the failure of the mechanical component will not cause trouble, since other mechanical devices will still operate and shut the reactor down if necessary. Only if two or three, or in important cases even more, major failures occur simultaneously is there even a chance of an accident. Since each component has only a very small chance to fail, because of its careful construction, the simultaneous failure of several independent components is almost unbelievably small. We have, as it were, in each situation several strong and independent lines of defense.

As I have said, the hazard of nuclear reactors is due to the radioactive substances in it. Their presence arises as follows. In the fission process the uranium nucleus splits into two smaller nuclei. Most of the resulting fission fragments are radioactive, that is, they change into still other nuclei in the course of time and in doing so they emit radioactive rays. The hazard potential in a reactor is, therefore, [fol. 3962] determined by the amount of fission products which are contained in the reactor. This amount depends primarily upon the power of the reactor. The reactor which PRDC proposes to build has only moderately high power, 100,000 kilowatts of electrical power or 300,000 kilowatts of heat. At least one other reactor whose construction has been approved will have more than twice this power.

The amount of fission products in a reactor does not depend on the mass of U^{235} in the reactor. This mass is indeed high in a fast reactor. In the PRDC reactor, for instance, it will be about 485 kilograms or a little over 1,000 lbs., which is about twice as much as in a thermal reactor of similar power working with enriched and solid fuel. However, this is compensated (for the present consideration) by the fact that at any given time there are fewer radio-

active fission product nuclei per pound of U^{235} in the fast reactor. In fact, if a reactor is operated for a very long time—many years—always at the same power, and if none of the fission products are removed, then the total amount of radioactivity in the reactor will depend only on the power at which it has been operated, and will be proportional to that power. Even the distribution of fission products among various chemical species is nearly the same for a fast as for a thermal reactor, and the small existing differences have no influence on the hazard.

Actually, the fuel elements from a reactor are unloaded periodically, and are reprocessed chemically to remove the fission products. The used-up fuel elements are replaced by fresh ones in a reloading process. In a fast reactor, the reloading period is usually relatively short; in the PRDC reactor, for example, it is planned to reload and replace some of the fuel elements every week, and any given fuel element will stay in the reactor only about 12 weeks. This means that some of the long-lived fission products will be [fol. 3963] present in much smaller amounts than in a reactor which is reloaded less frequently, as is often the practice in thermal reactors. These long-lived fission products, and especially strontium-90 which is among them, are generally considered particularly harmful biologically, and keeping their concentration down will reduce possible hazards. Thus, the PRDC reactor will actually contain slightly lesser amounts of dangerous fission products than a typical thermal reactor of the same power. But the main point is that a fast reactor does not contain any more fission products, or any more hazardous fission products, than a thermal one.

In normal operation of any reactor the fission products are enclosed within the reactor and cannot escape. Their radiation cannot get out even into the building surrounding the reactor, let alone to the surrounding country. This protection from radiation is generally accomplished by a shield made of concrete and iron or lead or other material which is especially designed to prevent radiation from escaping. A large amount of knowledge exists on shield design, and substantial protection can be afforded by a properly designed shield. In my opinion, the shield prepared

for the PRDC reactor is conservatively designed and no fundamental question exists as to its functional safety.

The only hazard, then, could come from a release of the fission products from their normal confined location in the uranium inside the reactor. Protection against this can be provided by adequate containment; fission products can be released only if the several containing vessels which usually surround the reactor were to break in succession. For example, in the proposed PRDC design, as disclosed in the License Application as amended, the reactor itself is enclosed in the reactor vessel, a strong steel vessel which will stand up under considerable pressure. Outside this is the graphite primary neutron shield which is surrounded by [fol. 3964] the primary shield tank, again of steel. A heavy concrete and steel biological shield surrounds the primary shield tank and the entire system is located in the reactor compartment, a sealed region filled with an inert gas. The reactor compartment is below the operating floor of the reactor building, and the outside wall of this building consists entirely of a strong steel shell.

The primary containment is by the reactor vessel, and every precaution is taken to design this vessel with the greatest care. Precautions are taken against a large number of conceivable causes for its mechanical failure. One of these might be heat shock, i.e., sudden change of temperature of the sodium coolant; elaborate precautions are taken against this affecting the vessel. Another cause of trouble might be chemical action between sodium and water; strong safeguards are taken against this. Mechanical shock might be still another and this also is guarded against.

Of course, none of these various conceivable causes of failure of the reactor vessel is peculiar to a fast reactor. Heat shocks may occur just as easily in water-cooled thermal reactors as in fast reactors. Also, sodium cooling is not peculiar to fast reactors but has been used in intermediate-energy ones and is being incorporated in some thermal reactors, due to its great advantages for heat transfer. There is still another possible cause for failure of a reactor vessel, namely, high pressure inside it, combined with mechanical fatigue of the material of the vessel: many thermal reactors, especially the pressurized

water reactor, use high pressure, while fast reactors, including the PRDC reactor, do not. In short, the possible non-nuclear causes of failure are present in thermal as much as in fast reactors, and in some cases more so. Failure due to nuclear accident is in my opinion far less likely than any of the improbable non-nuclear events which I mentioned. Of course, nuclear accidents also can be conceived for ther- [fol. 3965] mal as well as for fast reactors; their discussion will form the main part of my testimony.

Even if the reactor vessel in a fast reactor of the type proposed by PRDC should be breached, the fission products will not be released to the outside. In the first place, the fission products are still enclosed in the fuel elements themselves. These fuel elements must lose their integrity, by melting, burning, or crumbling into small fragments, before the fission products can escape.

Some fuel elements, more or fewer, according to the type of accident, may, of course, disintegrate and release their fission products. But we must remember that most of the fission products are solid or liquid at normal temperatures. These liquids or solids will then merely be sprayed over the primary shield. Some fission products are gases. These, in case of a breach in the reactor vessel, will still be contained in the primary shield tank, and if this also fails, in the gas-tight reactor compartment. It would be very difficult for them to find their way up to the operating floor of the reactor building. If they did, or if the reactor compartment were to break also, the fission gases would still be in a confined space, namely the reactor building.

To prevent them from escaping from this building, we have the last and strongest line of defense, the steel shell of the reactor building itself. This steel shell is made gas tight and thus can contain the gaseous fission products. Therefore, even if the reactor vessel, the primary shield tank and the reactor compartment should be breached this will not cause a hazard to the surrounding country but will merely make the reactor building inaccessible for a number of weeks or months.

The gas-tight shell of the PRDC reactor building is made of steel so that it can withstand some pressure. This is to [fol. 3966] provide for the remote possibility that the

sodium in the reactor might burn if the reactor vessel breaks which would heat the air in the building and thus increase the pressure. Even if all the sodium were to burn, the steel shell is designed to contain the residual air and the fission gases.

[fol. 3967] (4) *Nuclear Accident.*

One of the improbable events which might release fission products from a reactor is a nuclear accident. This could come about if the power of the reactor were suddenly to exceed the normal power by a large amount. Due to the excess of power, the fuel elements would heat up; they might then melt and the coolant might vaporize.

Such nuclear accidents can come about by a malfunctioning of the reactor control. As I mentioned before, a reactor has control rods which are moved mechanically and which determine the multiplication factor k. If the control rods are set in such a way that k is greater than one, then the number of neutrons and the reactor power will increase and if this increase is not stopped, an accident could occur.

There are many safeguards built into all reactors so that this cannot happen. Perhaps the most important is the power scram mechanism. When the power exceeds a certain level which is previously determined, then an automatic device will shut down the reactor. That is, the device will cause certain control rods to move rapidly into the reactor. This, as we have said, will decrease the multiplication factor k below one and thus lead to a quick drop in the reactor power.

These special control rods which can be rapidly pushed into the reactor are called safety rods. In the PRDC reactor, for example, they will be normally suspended above the reactor core by means of an electromagnetic clutch. When the scram signal comes, the magnetic clutch is released and the safety rods drop into the reactor. If for some reason the apparatus does not function properly, for instance, if there is a power failure, the magnetic clutch automatically ceases to operate, the safety rods drop into the reactor and the reactor is shut down. In other words, [fol. 3968] any failure in the system will lead to increased safety, the safety rods fail safe.

The power scram mechanism on a reactor is set at a power level which may be some 20% higher than the normal operating level, and some 20 to 50% lower than the power level at which fuel elements would melt. Safety rods are, of course, frequently tested to insure their continued safe operation.

Why is it necessary to ever let the multiplication factor k become greater than unity? We must permit such excess reactivity (k greater than one) in order to get the reactor started. When the reactor has been shut down for any reason, its power has to increase again from a low value to the operating level. This requires reactivity greater than one.

The higher the excess reactivity k — 1, the faster will the power of a reactor increase. The rate of increase of power is measured by the time which would be necessary to have the reactor power increased by a factor of about 2.7; this we call the period of the reactor. When the reactor is started up, this period is in general kept at a minute or more by suitable setting of the controls, but it may also be hours or it may be as short as seconds. It is a possible hazard to let the reactor power increase too fast because we might thereby lose control. For this reason another safety device is ordinarily provided on reactors which prevents the period from becoming too short. For instance, we may decide that we shall never operate at a reactor period shorter than ten seconds. In this case, a second scram device is provided which shuts the reactor down if its period becomes shorter than ten seconds. This is called the period scram in contrast to the power scram discussed before.

Properly designed scram devices have always worked in the history of reactors. Even in the few instances when small accidents have occurred in reactors, such as in the accident of the experimental breeder reactor EBR-I of [fol. 3969] the Argonne National Laboratory, the power scram apparently worked. However, in this EBR-I accident, the period scram had purposely been disconnected in order to perform a certain experiment which will be discussed in more detail later on. For this reason, the period was permitted to become as short as one second, correspond-

ing to a value of k considerably in excess of one. Now there is necessarily a delay between the power scram signal and the actual movement of control rods which substantially reduces the reactivity of the reactor. This delay is usually a second or so, and if the period itself is as short as a second, then the power will have increased considerably beyond the scram power by the time the reactor is actually shut down. For this reason, the EBR-I overheated and its core was partly melted, as described in more detail by Dr. Hilberry in his testimony.

However, if there is a period scram set e.g. at ten seconds, such an overshoot of the power cannot happen. Power will then never rise very fast, and during the delay from the scram signal to the actual shutting down of the reactor, the power will not rise appreciably. This hazard can be further minimized by making the delay from signal to shutdown as short as possible. For example, in any power reactor the period scram will be set at such a period that it will be entirely safe from overshoot of the power. Power reactor operators will never do any experiment in which they eliminate the period scram. The PRDC reactor, for example, will have a safe power scram set at a safe power level, and a period scram set at a safe period, and this combination seems to me to provide very real assurance of safe operation.

A further safeguard in the control of reactors which is incorporated in the PRDC reactor is that the control rods [fol. 3970] which are used by the operator to control the reactor power, as well as the safety rods, can be removed from the reactor only very slowly. Thus reactivity can increase only very slowly. This is in contrast to rapid insertion of the safety and control rods which can *decrease* the reactivity very rapidly. Mechanical means thus insure that all motions which lead to higher power and thus to hazard will be carried out slowly, while all motions leading to greater safety take place very fast once they are started. Slow increase in reactivity has the advantage that the measuring instruments can follow the power and the reactor period very easily and that there can be no overshoot.

There is no difference between the control of a fast reactor and the control of a slow reactor. Perhaps dif-

ferent materials and a higher volumetric concentration of poison may be used in a fast reactor than in a thermal one. However, the reactor periods which are involved are the same in both cases; they are of the order of seconds or minutes, and do not depend on the speed of the neutrons.

The reason for this is that some of the neutrons that are emitted in fission do not come out of the nucleus immediately but are delayed by times varying from about a second to about a minute. The control of a reactor, either thermal or fast, depends essentially on these delayed neutrons. In the case of U^{235} about .75% of all the neutrons emitted in fission are delayed.

If the multiplication factor k is exactly one, the chain reaction will be self-sustaining, but this will be true only if we use all the neutrons, including all the delayed ones. If we raise the multiplication factor k above one, we need less and less of the delayed neutrons. Finally, if k becomes about 1.0075 the delayed neutrons are no longer needed. For in this case it suffices to use 99¼% of the neutrons, those which are emitted promptly, and we shall still get as many neutrons in the next generation as in the last, and [fol. 3971] thus be able to sustain the chain reaction. Thus, for k=1.0075, a reactor is critical with prompt neutrons alone, and this condition is called prompt critical.

It is imperative to keep a reactor always below prompt critical, and reactor controls must operate chiefly in the region between the ordinary critical condition, called delayed critical, and prompt critical. For convenience, reactor scientists have called the interval from delayed critical to prompt critical one dollar. This dollar is used as the unit of reactiv. y and corresponds for U^{235} fuel to an interval of 0.0075 in k. This interval is subdivided into a hundred "cents". Thus a reactivity of 10 cents means a multiplication facter of k of 1.00075. Ordinarily reactor controls are set at no more than a few cents of reactivity.

There is a definite relation between reactivity in cents and the reactor period. For instance, one cent of reactivity corresponds to a reactor period of about 20 minutes. In general, this relation is expressed by a rather complicated mathematical formula known as the in-hour equation. This equation depends almost entirely on the prop-

erties of the delayed neutrons and is very nearly independent of the speed of the neutrons which cause the fissions. This statement is accurate to a small fraction of one percent at the low excess reactivities at which reactors are normally operated. But even if the reactor period were to get as short as one second, the corresponding excess reactivity is 75.5 cents for a fast reactor, and 76.0 cents for a rather slow thermal reactor. For this reason reactor control is the same for fast and for thermal reactors.

Only if a reactor were permitted to operate very near to or above prompt critical would there be any influence of the speed of the neutrons. In this case indeed the power in a fast reactor would increase more rapidly with time than [fol. 3972] that of a thermal reactor. However, no reactor is ever operated above prompt critical. In the operation of the PRDC reactor, excess reactivity will be limited at the most to about 70 cents, and will never be allowed to even approach the dollar which would make the reactor prompt critical and thereby hazardous.

As a matter of fact, it is perfectly feasible to design a reactor in such a way, and to load it with fuel in such manner that it will be impossible for the operator to increase the reactivity to a full dollar or close to it even if he were to forget all rules of operation. This safeguard can be applied in addition to period scram and power level scram, and, in fact it will be incorporated in the PRDC design. We simply shall not have a dollar excess reactivity available.

We are able to incorporate this safety feature because of the special properties of a fast reactor. The main property is that the fuel in the reactor burns up very slowly, because the critical mass is so large and the power per unit mass not particularly high. Now in the PRDC reactor, fuel will be reloaded every week, and between two such reloadings only about .5% of the fuel will burn. It is conservatively estimated that this burn-up, together with all accompanying changes like the accumulation of fission product poisons, will consume less than 42 cents of reactivity. It is planned to provide an additional 30 cents as a control margin for the operator. This means that even directly after reloading no more than 72 cents of excess reactivity

are available anywhere in the reactor once it is at the operating temperature even if all the control rods and safety rods are withdrawn. As I have shown, up to 72 cents and somewhat higher, there is no appreciable difference between the behavior of a fast and of a thermal reactor. No con-[fol. 3973] ceivable failure of either the mechanical devices or the operator can bring the reactor to prompt critical once it is operating after completion of a reloading.

Thermal reactors require far greater provisions of excess reactivity. This is because the fuel in these reactors burns relatively faster since the total amount of fuel initially provided is lower than that in a typical fast reactor. Hence at the same power level a thermal reactor has a greater rate of reactivity change due to fuel burn-up than a fast reactor.

More serious than this fact for the control of thermal reactors is the nuclide Xe-135. This nuclide is a fission product which has an enormous capture cross section for thermal neutrons, but almost none for fast neutrons. In a reactor operating at high power, large amounts of this nuclide will be present which will greatly decrease the reactivity. On the other hand, before the reactor has operated, Xe-135 will not be present. Therefore, in this "clean" state a large amount of excess reactivity has to be provided so that it can then gradually be given to the reactor as the xenon-135 builds up. For thermal reactors of high power, ten dollars or more of excess reactivity must be available for this xenon over-ride. No nuclide of similar properties is expected to exist in the case of fast reactors; for these reactors, all nuclei will have more or less the same cross section, and unusually high cross sections are impossible by the laws of nuclear physics. Therefore, it will not be necessary in a fast reactor to provide large excess reactivity when the reactor is clean for the purpose of overriding the absorption of some nuclide when the reactor is filled with fission products. From this very important standpoint then, a fast reactor can be made safer than a slow one.

[fol. 3974] (5) *Special Hazards in Fast Reactors*

I believe, therefore, that a fast reactor of the general type proposed by PRDC can be designed and operated

safely. However, there have been some adverse statements made about the safety of this reactor type, and I believe such statements are based on certain nuclear characteristics of a fast reactor which are not found in a thermal reactor. There are three main characteristics of this type.

1. Probably the most important of these lies in the fact that if ever the reactivity becomes greater than prompt critical, the power in a fast reactor will rise with a faster period than in a thermal one. I have explained before that this is not the case as long as the reactor stays below prompt critical, and that many independent precautions are taken to assure that a reactor never gets beyond prompt critical or even near it.

Although I see no way in which prompt critical can ever be exceeded in a reactor such as that proposed by PRDC, we are nevertheless, prepared to consider what could happen if through a completely unforeseen combination of circumstances, we should exceed prompt critical.

In this case indeed a fast reactor would heat up more rapidly than a thermal one, and a nuclear accident would occur. However, even if prompt critical were exceeded this does not result in an explosion which in any way would be comparable to an atomic bomb. In order to obtain an atomic explosion, one has to raise the reactivity of the fissionable materials in the atomic bomb very suddenly to a high value; in fact a reactivity of perhaps one hundred dollars has to be reached in a few millionths of a second. Moreover, even then a bomb explosion will not result if a large number of neutrons is present to begin with, as will be the case in an [fol. 3975] operating reactor. In short, it is extremely difficult and requires special designs to make an atomic bomb, and one cannot obtain an atomic bomb by accident. The high level of neutrons in a reactor is in itself an absolute guarantee that no atomic bomb explosion can ever result.

If a reactor ever has an accident, this can be, at the worst, comparable to a boiler explosion in a conventional power plant. Its severity depends primarily on the *rate* at which the reactivity increases beyond prompt critical. Now, all the motions of control rods in reactors are designed in such a way that reactivity can increase only at a very slow rate. For example, in the PRDC design, an increase of 1¢

per second is the most that the mechanism will permit. APDA has carried out calculations with which I am familiar, to determine the course of the accident which would result if all scram mechanisms were to fail and reactivity were to be inserted at the rate of 1¢ per second, even beyond prompt critical. These calculations have shown that in this case the natural thermal expansion of the uranium in the reactor would automatically shut down the reactor even though prompt critical was exceeded. In other words, even if all shut down mechanisms were to fail, the natural behavior of the uranium when heated would prevent an accident. We would merely get some rise in temperature which would subsequently die down.

Another calculation with which I am familiar was made in which the reactivity was allowed to increase 1000 times faster, at $10 per second; although I cannot conceive of any movement of the controls which would make the reactivity increase at anything near that rate. This calculation was carried out by Nuclear Development Corporation (NDA) at White Plains, N. Y., a nuclear consultant to APDA. The result of this calculation was that even this rapid increase of reactivity would not lead to serious trouble, and would [fol. 3976] presumably not burst the reactor vessel.

2. The second special nuclear characteristic found in fast reactors is that their reactivity would increase if moderators were introduced into the reactor. For instance, if sodium in a fast reactor coolant system were partially replaced by some material containing hydrogen—for instance by oil or water—the reactivity would increase. If this were done to an appreciable extent, the reactivity might increase beyond prompt critical. Exclusion of moderators from a fast reactor system does not, however, present any insoluble problems. In the PRDC reactor, for example, there is no conceivable way in which oil or water might get into the sodium system which is completely sealed. To make doubly sure, water is to be completely excluded from the reactor building—there are no drinking fountains, no wash-stands, nothing else containing water in the building. Oil and grease are kept to an absolute minimum; there are no reservoirs of lubricants as are common with large motors, but each piece of equipment which requires it is lubricated only by a small cartridge of lubricant.

Thus only very small amounts of hydrogeneous material are available anywhere in the building, and these small amounts cannot get into the sodium system. Nevertheless, NDA has also carried out calculations with which I am familiar about the effect of substituting moderator for sodium in the PRDC reactor. Even making unduly pessimistic assumptions, no serious accident resulted unless an immense amount of moderator was introduced, and I consider this incredible.

I am acquainted also with what I consider to be a somewhat more realistic calculation, made by APDA, which envisaged the possibility that suddenly the sodium entering the reactor was 100 degrees cooler than the sodium normally entering. However this should come about, I cannot con- [fol. 3977] ceive, but this possibility again did not lead to a nuclear accident.

3. A third relevant characteristic of fast reactors is the large amount of rather highly enriched fissionable material contained in them. If all this material in the PRDC reactor, for example, were assembled in one solid ball, and then surrounded tightly by a good neutron reflector such as solid natural uranium, we would have about six times a critical mass. If brought together suddenly, such a large amount of material would explode—but as I have pointed out already, with nothing remotely resembling the force associated with an atomic bomb. Thermal reactors of equal power working on enriched U^{235} contain less U^{235} and therefore do not involve this hazard, especially if the U^{235} is not highly enriched. In fast reactors, only the geometric dispersion of the fuel for cooling purposes prevents the reactor from being supercritical. For this reason, it is essential to make sure that the fuel in a fast reactor can never assemble in one large mass. It is feared that such an assembly might result if a fast reactor melted down. At present, APDA, Argonne, and others are conducting very thorough investigations to make sure that any conceivable melt-down will not lead to the assembly of a large volume of enriched uranium.

Of course, every precaution is taken in any well-designed fast reactor (such as the PRDC reactor) so that even the

first part of this hypothetical accident, namely, the melting down of the uranium, will never occur. I have discussed some of these precautions before and will discuss others later on. However, it is argued, the experimental breeder reactor EBR-I *did* partially melt down, in an incident in November, 1955, and this incident has caused most of the fears concerning fast reactors. It is my sincere belief that these fears are groundless and that the causes which made the EBR-I melt down will not exist in the PRDC reactor. [fol. 3978] This is because the structure of the PRDC reactor will be different from EBR-I, and the reactor will be operated in a different manner. I shall discuss these differences further in this testimony.

4. Finally, a fourth characteristic relevant to safety is sometimes attributed to fast reactors and not to slow ones as a class. This is the observation that EBR-I had a positive temperature coefficient of reactivity which was the immediate cause of its accident and it is sometimes argued that a positive temperature coefficient might be a general feature of fast reactors. I do not believe that this is the case, and I shall give my reasons for this opinion in the next section in which I shall also explain the meaning of the term positive temperature coefficient.

The positive temperature coefficient and the consequences of a possible melt down of a fast reactor are problems which do require further experimental and theoretical study. In the next three sections of this testimony, I shall deal with the studies which have already been made on these problems, and with those which are now being conducted and are planned for the coming months. In my opinion, this work will establish that coefficients of reactivity and melt down problems in fast reactors do not render this type of reactor hazardous for operation in populated areas.

To summarize, there are many safety features of reactors. They include well-trained operators, good instrumentation, many independent automatic shut-down mechanisms, control systems designed to fail safe, a sealed-up coolant system, and many independent, strong containing vessels. There are some features peculiar to fast reactors which tend to make them even safer than thermal ones, e.g. that the system is at low pressure, that the reloading cycle of the

fuel is short, and especially that only a very small amount of excess reactivity needs to be provided because there is no [fol. 3979] need for xenon override. Taken all together, it is my opinion that on the basis of present knowledge there is as much assurance of designing and operating a fast reactor safely in a populated area as there is of designing and operating a thermal reactor, with the sole possible exception of the two areas of temperature coefficient and melt down which are still being examined both theoretically and experimentally. It is my further opinion, that a fast reactor can be so designed and constructed as to provide adequate safeguards in these two areas, and that the theoretical and experimental work now underway in these fields will confirm this.

(6) *Temperature Coefficient*

It is generally true that an increase in the temperature of a reactor will lead to an expansion of the fuel and of the rest of the reactor, and that this expansion will decrease the reactivity. If this is the case, we say that the reactivity has a negative temperature coefficient. In calculating the effect of mild accidents, e.g. in the calculations of APDA and NDA which I mentioned before, a negative temperature coefficient is generally assumed and its value is calculated from the known thermal expansion of uranium and the other constituents of the reactor. If the normal mechanical safety devices of a reactor should fail, the negative temperature coefficient is the chief mechanism which prevents severe accidents.

Unfortunately, the EBR-I accident in November, 1955, which I mentioned before has shown that the EBR-I reactor had a positive temperature coefficient. The power and the reactivity were carefully recorded during the course of the accident and there is no doubt that the reactivity of EBR-I increased as the reactor heated up. In fact, it had been known for more than two years before the accident that EBR-I had a positive temperature coefficient. Clearly, the positive coefficient was the direct cause of the accident [fol. 3980] of November, 1955: as the power increased the reactivity of the reactor increased, which then led to an even faster rise of power, and so on.

Obviously a positive temperature coefficient cannot be tolerated. Its existence had given grave concern to the scientists at ANL, APDA and NDA ever since it was discovered, and the experiment of November, 1955, which led to the accident was part of an extended program to find out more details about the positive temperature coefficient and, if possible, to determine its cause. A further program for investigating its cause and possible cure is well under way.

It is necessary to qualify the statement that the EBR-I had a positive temperature coefficient. Actually, if the *whole* reactor was heated slowly, the reactivity of EBR-I decreased as it does for all reactors I know. The positive coefficient was observed only if the fuel was heated very rapidly so that there was no time for the sodium coolant to carry the heat away. We therefore say that the *prompt* temperature coefficient is positive; on the other hand, if one waits long enough, the total temperature coefficient is negative. Therefore EBR-I behaved all right in normal operation. Indeed, EBR-I was operated for four years without any trouble, and many experimental results were obtained from it. Only when a special experiment was made in which the operating conditions were entirely different from those in normal operation and in which the reactor was purposely put on a fast rising period by inserting considerable excess reactivity, then the prompt coefficient came into play which finally led to the accident.

The fact that in this case we have to do with a prompt temperature coefficient shows immediately that this coefficient must be associated with the temperature of the fuel itself. The question arises now whether the positive [Fol. 3981] temperature coefficient is in any way peculiar to fast reactors, and whether perhaps it is unavoidable in fast reactors. I am confident that this is not the case.

It is true that an effect exists which *could* give a positive contribution to the temperature coefficient in EBR-I and which *is* characteristic of fast reactors: This is the so-called Doppler effect. Investigations on this effect have proceeded for almost ten years, mostly at the Knolls Atomic Power Laboratory at Schenectady. However, that Laboratory was primarily interested in intermediate reactors in which the neutrons are neither thermal nor fast. For fast

reactors, the first thorough theoretical treatment of the Doppler effect was given two years ago by Dr. Goertzel and his associates at NDA. I am familiar with their calculations and in fact followed this work very closely due to its extreme importance. Their main results were, first, that the temperature coefficient due to Doppler effect is very small and second, that it depends critically on the dilution of U^{235} with U^{238}. In EBR-I in which nearly pure U^{235} is used, the Doppler effect may indeed cause a small positive temperature coefficient. But if U^{235} is diluted with at least one atom of U^{238} per atom of 235, then the Doppler effect will give a negative temperature coefficient. In the PRDC reactor, for example, there will be about 3 atoms of 238 per atom of 235, and this will make the Doppler coefficient certainly negative and its value somewhat larger than for EBR-I. Therefore, according to the best theoretical information now available, the Doppler effect can be made to contribute to safety in fast power reactors and will do so in the PRDC reactor.

Even in the EBR-I, the Doppler effect cannot have been the cause of the observed positive temperature coefficient. [fol. 3982] According to my own latest calculations, it could have contributed no more than 5% of the observed positive temperature coefficient.

Experiments are under way at Arco, Idaho, to measure the Doppler coefficient in a critical assembly, ZPR-3, which contains the same materials as EBR-I. A critical assembly is a reactor which operates at extremely low power, perhaps a few watts, a condition which is particularly suitable for exact measurements of the reactivity. The experiments so far carried on have given the result that the Doppler effect may be even smaller than the calculated value, certainly not larger. These difficult experiments are continuing, and are discussed by Dr. Hilberry in his testimony. At present they give confidence in the theoretical calculations and confirm the conclusion that the Doppler effect cannot have been responsible for the positive temperature coefficient in EBR-I; and will certainly not give a positive temperature coefficient in the PRDC reactor.

The experiments prove in fact somewhat more. The experimental fact is that the heating of a piece of U^{235}

by itself, without heating the rest of the reactor, does not cause a measurable increase in reactivity, as long as the uranium does not move. This excludes not only an undesirable Doppler effect but means that there is also no other internal effect of any kind in the fuel which gives an appreciable positive temperature coefficient. Thus the experiments show that we have not overlooked any physical phenomenon which might give a positive temperature coefficient. Thus we have direct assurance that there is no intrinsic property of fast reactors which causes a positive temperature coefficient.

It is therefore generally agreed that the positive temperature coefficient observed in EBR-I was due to some features [fol. 3983] in the mechanical design of this reactor rather than to the high speed of the neutrons. Now the EBR-I was designed to make certain physical measurements, in particular to establish that breeding is possible. It was not intended for engineering ~~tests,~~ and was therefore not designed with the care, and did not have the mechanical rigidity, that one should demand of a power reactor.

The fuel elements in EBR-I are rods of U^{235} which are held at their top and bottom. When the reactor operates, the side of each fuel rod which is closer to the center of the reactor will become hotter than the side away from the center. This will make the inner side expand relative to the outer, and the fuel elements will bow. Since its top and bottom are fixed in vertical position, the middle of the fuel element will be pushed closer to the center of the reactor. This will lead to an increase of reactivity.

It is rather generally believed that this was the cause of the positive temperature coefficient in EBR-I. One can estimate the amount of bowing that should occur at a given reactor power, and this is just about the right amount to explain the observed positive temperature coefficient. Unfortunately, it is not possible to make measurements of the position of fuel elements in the reactor while it is in operation, and therefore, we have no positive proof that bowing of the estimated amount actually took place in the reactor.

To find out whether bowing was responsible a program is at present under way to rebuild the EBR-I with a different mechanical design, as explained in more detail in the

testimony of Dr. Hilberry and Mr. Davis. The fuel elements will be made more rigid and will be held rigidly in place so that bowing will be greatly reduced. This program is being carried out at Arco, Idaho, by personnel of the Argonne National Laboratory. APDA and PRDC are vitally interested in the program and are helping to the best of their [fol. 3984] abilities.

In the design of the PRDC reactor, for example, the engineers have been aware for some time that inward bowing of the fuel elements must be avoided, and several alternative methods are being examined. The testimony of Mr. McCarthy will describe these designs more in detail. The intention is to assure either that no bowing occurs at all or that bowing should push the fuel elements farther apart in the core rather than closer together, and if this can be achieved, the reactivity will decrease rather than increase due to bowing.

The outcome of the test of the new EBR-I design will greatly influence the further design of fast power reactors, including the PRDC reactor. If it is found that the positive temperature coefficient has been removed as I am confident will be the case, we shall feel justified in going ahead on the basis of present design criteria; if it is not, we shall make further changes in design and further experiments to insure a negative temperature coefficient. The EBR-I test is scheduled to take place in the summer of 1957.

If bowing can be made to contribute a negative temperature coefficient, or if it can be eliminated, the total prompt temperature coefficient of the PRDC reactor will be negative, because of the thermal expansion of the fuel elements. With a negative prompt temperature coefficient, an accident of the EBR-I type cannot occur, the reactor will shut itself down when heated even if the heating occurs rapidly, and the reactor will then be safe.

The EBR-I experiments also revealed another type of instability, a resonance. When the reactor was operated at [fol. 3985] its normal power and the flow of sodium was reduced, a point was reached where the reactor power would begin to oscillate with a period of about fifteen seconds, without any external cause. These oscillations were investigated in considerable detail by means of a re-

actor oscillator, a device which I shall describe in more detail later in this testimony. The resonance is somewhat but not too closely related to the positive temperature coefficient. The resonance does not constitute a direct hazard but it would limit the power at which the reactor could be operated.

I have every reasonable expectation that the projected work on EBR-I will isolate the causes of instability, both with respect to temperature coefficient and resonance, and there is no reason to expect that any real difficulty will be encountered in designing a fast stable reactor with negative temperature coefficient of adequate magnitude and sign. I further expect that the experimental program will show the design of the PRDC reactor to be stable; at the least, I am confident that it will indicate design changes which can be made to accomplish stability.

(7) *Meltdown in the Presence of Sodium*

In normal operation all known reactors are extremely stable. Even the EBR-I which had a positive prompt temperature coefficient operated with great stability and reliability for four years. Normal operation will, therefore, not lead to meltdown with its possible consequences.

Meltdown is conceivable when a radical change has been made in the composition of the reactor. One major change which occurs regularly is the reloading of the reactor with new fuel elements. Investigations have made whether and in which way meltdown could occur in the PRDC reactor after reloading.

[fol. 3986] Of course, reloading is conducted in such a way that meltdown will actually not occur. The new fuel elements will contain just as much U^{235} as the old fuel elements did when they were fresh. As a precaution, care is taken that the new fuel elements are not inserted in the center of the reactor where they would contribute most to the reactivity but only in the outer parts of the core where the reactivity contribution is much smaller. The used-up central fuel elements will in turn be replaced by partly-used fuel elements which were previously located in the outer parts of the reactor core. This procedure will minimize any possible excess reactivity introduced by reloading.

The main concern then would be that perhaps one fuel element might slip in which contains U^{235} of a higher enrichment than is required for the reactor. It is not clear how this could come about, but to make doubly sure, APDA is at present considering testing procedures to which each fuel element pin would be subjected before insertion into a sub-assembly. This testing would be directed toward insurance that each fuel pin has exactly the U^{235} enrichment required.

Let us assume that in spite of these precautions, a fuel sub-assembly is inserted which is too highly enriched and therefore gives a larger contribution to reactivity than required. If this were to happen, it would be immediately discovered even while the reactor is being reloaded, because the instruments would indicate a rise to a higher neutron level. The instruments which are used in the shut-down condition during a reloading are sensitive enough to detect a reactivity which exceeds the required one by a fraction of a dollar. If such extra reactivity is discovered, the reloading will be immediately interrupted, some of the freshly loaded fuel element sub-assemblies taken out and replaced by others until the instruments indicate the correct [fol. 3987] reactivity.

Let us assume, however, that the operators responsible for the reloading (who of course will be particularly competent) fail to discover the extra reactivity. Then after the reloading process is completed the safety rods will be withdrawn and thereby the reactivity allowed to approach unity. As I have pointed out before, this will be done very slowly, at a rate not exceeding 1¢ per second, and the instruments will be operating all the time. If the neutron detectors do not work properly, an interlock will prevent the withdrawal of the safety rods. Assume now that the reactor has been loaded with too much fuel, that the operators have failed to notice this and that the safety rods are withdrawn. Then as soon as the reactor period becomes shorter than the setting of the period trip, the scram will operate and the reactor will be shut down. The same would happen if the power exceeded the operating power. Therefore, no meltdown will result even if the wrong type of fuel

element has slipped through the fuel element testing and even if the operators have failed to notice this.

An excess of power can only result if, in addition to these two unlikely events, there is a complete failure of the safety mechanism, if for instance the safety rods which were just withdrawn cannot fall back into the reactor. It is quite inconceivable to me just how this could happen. If it were to happen, and if in addition the movement of the safety rods out of the reactor could not be stopped—another most unlikely assumption—then the reactor might overheat and melt down.

These supposed conditions of a reactor meltdown, and other even more unlikely chains of events, would be quite similar to the conditions which were purposely created for experimental purposes when the EBR-I meltdown occurred. The postulated conditions would be somewhat worse than [fol. 3988] those under which the EBR-I accident occurred, however, because it is postulated that none of the safety mechanisms would work.

Let us, therefore, have a closer look into the meltdown incident of the EBR-I. The Argonne National Laboratory, as Dr. Hilberry has explained in his testimony, has studied in great detail the conditions of the core of EBR-I after its melting, and I am familiar with these studies; in fact, I have also examined the EBR-I core after meltdown. They have found that uranium was expelled from the center of the core. The density of the core before the incident was about 10.5 grams per cubic centimeter; after the incident the density at the center of the core was reduced to about four grams per cubic centimeter, and this remaining material was very porous. Apparently more than half of the uranium which was originally at the center of the core had been pushed out by the melting to a position near the edge of the core. Indeed, both on the sides of the reactor core and near its lower edge, one finds material which apparently was pushed into these positions from somewhere else, in addition to the material which was in these positions from the beginning and which shows no signs of melting. Below the center of the core there is a region in which the density was increased to 14 or 15 grams per cubic centimeter by addition of extra material. Thus

there is ample evidence that material from the center of the core was pushed toward the outside.

Now it is well known that the contribution of U^{235} to the reactivity is greatest when the U^{235} is near the center of the assembly. Material near the edge of the core contributes far less to the reactivity, and material pushed into the blanket would contribute less still. Therefore, the meltdown incident of EBR-I, far from assembling the material into one big mass of uranium, disassembled it and thereby reduced the reactivity.

[fol. 3989] While there is no positive evidence on the mechanism of this disassembly, it is reasonable to speculate that it might have happened like this. The molten uranium came in contact with liquid sodium, after eating its way through the steel cladding of the fuel elements. The boiling point of sodium is below the melting point of uranium; therefore the sodium immediately started to boil. Because the processes were rapid, the boiling sodium did not find an outlet along the normal coolant channels. Instead its pressure pushed the uranium away from the center of the core into the outer parts of the reactor.

Whether this mechanism is the correct explanation can only be determined by further examination of the EBR-I core, and by further theoretical work. However, it seems clear already that there was a rather strong force propelling the uranium outward. If the accident had been more severe, this force would have been still stronger and would have reduced the reactivity of the reactor even more.

It is not known whether the movement of U^{235} in fact shut down the EBR-I in its incident. The operator of the EBR-I hit the scram button when he noticed the fast power rise, and the automatic power scram was also functioning properly. It is therefore possible that the EBR-I was shut down by these scram mechanisms rather than by the expulsion of uranium from the center of the core. However, had all the scram mechanisms failed, it is clear in my opinion that the movement of fissionable material away from the center would itself have shut down the reactor.

Therefore, it appears that the melting of uranium in the presence of sodium which can vaporize does not lead to an

assembly of a supercritical mass but is instead a powerful mechanism for the shutdown of the reactor. At the same time, the pressures generated (presumably in the sodium [fol. 3990] vapor) are not likely to be enormously great. One evidence for this is the fact that gravity had time to operate in the EBR-I incident and assemble more uranium near the bottom than near the top of the reactor. Further, the forces were far too weak to burst the reactor vessel. Indeed, in the EBR-I incident, even the rather weak blanket was left intact.

Further careful analysis will be needed to establish all these points more reliably. However, even now it seems very likely that a meltdown in the presence of sodium will not bring the reactor above prompt critical but will provide an effective automatic shut-down mechanism.

[fol. 3991] (8) *Meltdown After Loss of Sodium*

Meltdown therefore can lead to serious trouble only if it occurs after the sodium from the reactor has been lost first. This is an even more unlikely eventuality than the startup accident which I discussed in the last section. Indeed, every precaution has been taken in the PRDC reactor design against the possibility of loss of sodium. Even if the pumps fail and if the power fails, the sodium will still remain in the system. As a matter of engineering, it is certainly possible so to design a reactor vessel and cooling system that the presence of sodium in the reactor will be guaranteed at all times, and I believe that the PRDC design as described in the License Application verbally does this. I believe that the sodium system of the PRDC reactor has been designed with greater care than that of any similar reactor, and that we can really be sure that a sudden loss of sodium will never occur.

Even if there should be a compound leak in the system, the sodium flows into a sump tank, and a sump pump takes it back into the reactor vessel tank. Thus, it has to be postulated that simultaneous with a compound leak in the system, there is a failure of the sump pump.

In this most unlikely contingency, the further course of events depends on the size of the leak. With any credible

leak, it will take many hours before the large sodium reserve tank is emptied of its 13,000 gallons of sodium. During these hours, the reactor can be unloaded beginning with the central fuel elements which contribute most to the reactivity. Even if this unloading cannot be completed before the complete loss of sodium, the remainder of the fuel in the periphery of the reactor is likely to be too dispersed to lead to a critical mass even if it melted down under the most unfavorable conditions.

[fol. 3992] If there were a loss of sodium through a leak of enormous proportions which persisted in spite of all the precautions taken such as double containment and the ability to recirculate, the sodium might run out of the large reserve tank in a few minutes. This would still give time for some precautionary measures, but I will consider the case when such measures are not taken, except that of course the safety rods are inserted and the reactor shut down immediately after the loss of sodium begins.

The reactor is then left after a few minutes without sodium and without power. However, the fission products accumulated from the prior operation of the reactor will continue to develop heat by radioactive decay. At a time of one second after shutdown, the fission product heat is about 6.5 per cent of the reactor power before shutdown. It then decreases as the 1/5 power of the time. Five minutes after shutdown, it amounts to about 2 per cent of the preshutdown power of the reactor.

In the absence of sodium, there is no way to dissipate this power. The reactor core will, therefore, heat up and melt, beginning perhaps half a minute after the sodium has been lost. The fuel at the center of the reactor will melt first, and then the melting will spread to the outer parts of the core in a time of another minute or so.

The details of the melting process are complicated and are only partially known at present. At the outset, therefore, it seemed desirable to try to avoid detailed consideration of the melting. To this end, two hypothetical situations were considered, both of them far worse than any credible situation. If these hypothetical situations had led to calculated accidents of moderate size which could have been safely contained in the reactor building, we would not have

needed to investigate in detail the probable course of a meltdown in the absence of sodium. Unfortunately, both [fol. 3993] calculations led to substantial release of nuclear energy; detailed investigation of the ~~meltdown~~ conditions has, therefore, become necessary, and is being undertaken both theoretically and experimentally by APDA and NDA.

The first study of an incredible maximum accident was made by NDA at the request of APDA. They assumed that after the loss of sodium, the sodium suddenly rushed in again, that it did so with the same speed it has in normal operation, and that in addition the sodium was able to move the (much heavier) molten uranium at the same great speed (30 feet per second). It is clear that these assumptions are little short of fantastic, and could not be realized under any conditions, not even if we purposely tried to do so. The energy release found by NDA was equivalent to 10,000 lbs. of TNT.

The second calculation of an incredible accident was done by Dr. Tait, a member of the British Atomic Research Establishment at Harwell, and myself. We did not assume that the sodium returned to the reactor. However, we assumed that all the uranium in the reactor melts without changing its position and then collapses under its own gravity. Although slightly more realistic than those of NDA, these assumptions clearly are still in the realm of fantasy. With these assumptions, I have calculated that a meltdown accident of the PRDC reactor would release an energy equivalent to about 1,000 lbs. of TNT. A more accurate calculation is still under way.

This figure I consider to be a gross overestimate. For instance, Dr. Tait and I assumed that the uranium does not fill the voids left by the loss of sodium, and this assumption, certainly incorrect, tends to overestimate the energy release, probably by a factor between 2 and 4. We assumed that all the uranium melts at the same time and then collapses all together; actually, the center of the reactor core will melt first and may flow away even before the outer parts of the core melt. Further, the uranium will trickle [fol. 3994] down rather than fall freely under gravity. All these and other considerations will tend to lower the energy

release very much. How much can only be determined by detailed investigation.

The original calculation by Tait and myself yielded an energy release of 350 lbs. of TNT rather than of the 1,000 lbs. which I now obtain. The difference between these two figures is mainly due to two causes. In the first place, the smaller figure referred to the British fast reactor which is being constructed at Dounreay, Scotland. The PRDC reactor has about twice as much uranium as the British reactor, and this increases the energy release in the postulated accident by a factor somewhat greater than the ratio of the masses.

Secondly, Dr. Tait and I assumed that no safety rods would be inserted into the reactor after the loss of sodium. Then the reactivity before collapse will be negative only because the sodium has been lost; this gives a negative reactivity of about $2. It is a sounder procedure to insert the safety rods as soon as the loss of sodium becomes apparent. If this is done, the reactivity is $10 negative, and it is very likely that no reasonable flow of uranium will bring the reactor back to critical again. I will discuss this further below, in connection with more realistic calculations of the accident which are at present being done at NDA. However, if we make the arbitrary and unrealistic assumptions of Tait and myself, namely that the accident nevertheless will happen in any case, and that the core collapses under gravity, we find the somewhat surprising result that the accident gets worse if the negative reactivity before collapse is greater. This is because a greater amount of collapse is required to compensate a great negative reactivity, and if a greater amount of collapse is assumed, a higher part of the core may be assumed to be included in the collapse and so to fall from a greater height and there- [fol. 3995] fore to acquire a greater speed due to gravity. This means that at the time when criticality is reached again, the reactivity increases at a faster rate if the safety rods have been inserted than without this insertion, and this in turn causes the energy release in the accident to increase.

An interesting result of the calculations of Tait and myself is that the energy release does not depend strongly

on the speed of the neutrons in the reactor. This speed is most conveniently measured by the neutron lifetime; i.e., the average time which a neutron takes, from its birth in one fission to its absorption in the next fission. The energy release is inversely proportional to about the fourth root of the lifetime. Since the lifetime in fast reactors is about 1/100 of that in thermal ones, the energy release in the Bethe-Tait type accident is only about three times what it would be in a thermal reactor. Thus, the influence of the "fast" character of the reactor is only moderate, even though prompt critical is exceeded in this type of accident.

It is not known whether an explosion of 1,000 lbs. of TNT equivalent can be contained in the many containers provided in the proposed PRDC reactor. The explosion would certainly burst the reactor vessel. Much of its violence would be dissipated in compressing the blanket and deforming the shield. Presumably some shock wave will still be transmitted into the air of the reactor building. Whether or not this shock wave can be contained in the steel shell of the reactor building is not known at present. If the answer is negative, it is again not known how much smaller the energy release has to be before it can be contained.

I understand that the Atomic Energy Commission is sponsoring an investigation of the whole containment problem. This investigation, which may include model experi- [fol. 3996] ments, may give us a much better idea on the limitations of containment. In principle of course, any explosion can be contained in a building of sufficient strength.

We should remember, however, that the model of Tait and myself is only an extreme assumption which has no relation to reality. The relatively large amount of explosive energy which was obtained in these calculations only indicates the necessity for a more detailed investigation of the actual sequence of events in the meltdown in the absence of sodium.

Such detailed investigations are now under way. As Dr. Hilberry has stated in his testimony, the Argonne National Laboratory is planning to construct a meltdown facility for the investigation of the actual process in the

melting of uranium fuel elements, and of the subsequent flow of the molten material down the reactor. ANL in collaboration with the Los Alamos Scientific Laboratory plans to take the results of ANL meltdown studies as the basis of calculations of the energy which might be released if a critical mass should be reassembled by the meltdown. The special knowledge of the Los Alamos Scientific Laboratory on nuclear explosions and on the behavior of matter at high temperatures will be very useful for these calculations.

APDA has also initiated similar studies of its own. Nuclear Metals, Inc., at Cambridge, Massachusetts, under the direction of Dr. A. R. Kaufmann, is setting up experiments to study the melting and flow of uranium, using similar geometric configurations to those which will exist in the reactor, but using natural uranium rather than U^{235}, and electric rather than nuclear heat. It is hoped that these studies will give detailed information on the melting of the zirconium cladding which is around the uranium fuel elements in the PRDC reactor, on the flow of molten uranium, and on the possible refreezing of this uranium.
[fol. 3997] At the same time NDA at the request of APDA is making theoretical studies of the same problem. One of their problems is whether the molten uranium will freeze again as it flows down the reactor. In principle, such refreezing is possible because the U^{235} in the lower part of the core contains less fission products than that in the center and, therefore, will not yet be up to the melting point when the center melts and begins to flow down. In the lower blanket section, below the reactor core, there is depleted uranium in which the fission product heat is negligible, so that the temperature is very low. The present tentative conclusion from the NDA calculations is that the molten uranium which flows from the center of the core will not freeze again in the lower section of the core but may at least partly refreeze when it comes into the lower blanket. These calculations are incomplete, and even after their completion we shall await confirmation of the theoretical results by the experiments at Nuclear Metals and Argonne National Laboratory.

The question whether the molten uranium freezes again while flowing through the bottom of the reactor is of prime importance for the further course of events. If there is no refreezing at all, then in the proposed PRDC reactor the material will leave the reactor region and drop down into the special meltdown compartment which is provided at the bottom of the reactor. In this meltdown compartment, the molten uranium is permitted to distribute itself over a very wide area, 8 feet in diameter, which will make the layer of uranium in this compartment very thin even if all the fuel should drop down into it. Such a thin layer will not be critical even if it is surrounded by good neutron reflectors. Detailed calculations will be made to insure that the full amount of uranium in the meltdown compartment will remain far below critical, taking into account reflection of the neutrons from the materials which may actually surround the fuel when it is in the meltdown compartment. For instance, it will be taken into account that the blanket [fol. 3998] may drop into the meltdown compartment in the course of events. Special care will also be taken to design the meltdown compartment in such a way that the fuel is certain to distribute itself uniformly over the area provided. If necessary, knobs or dividing walls containing a nuclear poison such as boron-10 will be provided in the meltdown compartment in order further to reduce the reactivity of the fuel when it is in that compartment.

If there is no refreezing, the flow of fuel through the reactor will be fast enough so that all of the fuel flows out very shortly after it melts. For instance, the fuel from the center of the reactor core will have flowed out long before the fuel near the edges begins to melt. In this way accumulation of fuel in any one section of the reactor, whether in the core or in the lower blanket, is minimized. I am confident that under these conditions the reactor will not get up to critical again if, by insertion of the safety rods, the reactivity starts out at $10 negative. NDA has started a more accurate calculation of the multiplication factor of the reactor as a function of time after melting begins. Once the flow of molten uranium is known, one can calculate the distribution of fuel at any instant of

time and can then calculate the value of the multiplication factor k for this distribution.

On the other hand, if uranium freezes again, for instance on its way through the lower blanket, the channels for its flow become clogged and further molten uranium will then accumulate on top of the clogged channels and mainly in the bottom of the reactor core. If this were to happen, conditions somewhat similar to those postulated in the Bethe-Tait accident calculations might arise. However, the rates of assembly of the fuel would undoubtedy be much slower than assumed in the Bethe-Tait calculation, first because not all the uranium would drop down at once and secondly because it would merely trickle down rather than [fol. 3999] fall freely by gravity. With the much smaller rate of assembly of the fuel would undoubtedly be much smaller and would appear to be containable in the many containers surrounding the reactor and finally in the reactor building.

However important it is to investigate the course of a possible meltdown accident with the present PRDC design, it is far more important that these same investigations may point the way to improvements in the design. It is very likely that no serious freezing of the uranium can occur in the lower part of the core but only in the lower blanket. If this is confirmed, and if the NDA calculations and the experiments at Nuclear Metals and ANL indicate a reasonable possibility that the uranium will refreeze in the lower blanket, APDA will seriously consider eliminating the lower blanket from the reactor. This will entail some moderate loss of breeding and some moderate increase of critical mass. It will, therefore, reduce the economic value of the reactor, though not too seriously. However, under the conditions outlined, the sacrifice of the lower blanket would eliminate entirely the nuclear accident after meltdown, even after the loss of sodium.

The possible removal of the lower blanket is only one example of positive action which can be taken in changing the reactor design depending on the results of the current research program. Other design changes may be made if called for. At present, we do not have sufficiently realistic information on the course of meltdown accidents, but only

educated guesses. However, a vigorous research program is under way to fill the gaps of our knowledge, and the design of the PRDC reactor is sufficiently flexible so that it can profit from the results of this program. I have reasonable hopes that this combination of research and design change will lead to a reactor which cannot have *any* nuclear accident after meltdown. I am confident that the [fol. 4000] reactor can be designed in such a way that no credible meltdown situation can result in an accident of sufficient energy to breach the containment building.

(9) *Pre-Operational Testing*

To the greatest extent possible, advance experiments and engineering evaluation are performed to assure the safety of any reactor, and this will, of course, be the case with the PRDC reactor. However, a reactor should be subjected to a rigid pre-operational testing which can be carried out safely at every step. Pre-operational testing is carried out following completion of the construction of a reactor. This procedure provides a method for further check on experimental and theoretical results, and to be most effective must be carried out on the reactor itself. Consequently, such tests cannot be conducted until a reactor is completely constructed and ready for the introduction of nuclear fuel.

Before the PRDC reactor is assembled in Michigan, its critical mass and many other nuclear properties will be measured in a critical assembly in the Arco, Idaho, facility of the Argonne National Laboratory. Because there are some necessary differences between the structure of a critical assembly and of an engineered reactor, it is expected that the critical mass of the final PRDC reactor may differ from that measured in the critical assembly by up to 5 per cent, either way. The reactivity value of the control and safety rods, and of fuel elements in various positions in the reactor, will also be measured in the Arco critical assembly. Moreover, it will be possible to measure the subcritical multiplication of a neutron source, with various percentages of a critical mass loaded in the reactor, and with various neutron detectors.

When the reactor is ready for assembly in Michigan, the fissionable material will be put in place piece by piece, in the presence of a strong neutron source. The details of the [fol. 4001] start-up procedure are described by Mr. McCarthy in his testimony. The central fuel elements are loaded first, and then the loading proceeds spirally outwards so that, as criticality is approached, the reactivity value of each of the last fuel elements is much less than a dollar (about 40 cents).

The subcritical multiplication of the neutron source by the fissionable material will be measured at each stage of loading with several neutron detectors of different characteristics. From these measurements, the critical mass can be predicted in various different ways. The most common method is to plot the reciprocal of the multiplication, 1/M, as a function of the mass of fuel. As the critical mass is approached, these plots become straight lines which point to the critical mass (at infinite multiplication M). Since several different neutron detectors are used, several different straight lines are obtained which should all extrapolate to the critical mass and thus provide a check on each other. This method, which has been used on many reactors in the past, permits a very accurate prediction of the critical mass.

Another method which is even more accurate, especially at early stages of the assembly, makes use of the previous experience with the Arco critical assembly. If the same neutron source and neutron detector are used, then the same multiplication will be measured in the PRDC reactor and in the Arco critical assembly, when the same fraction of the critical mass has been assembled. Thus, it will be possible to determine the fraction of critical mass assembled at every step.

The last steps before criticality is reached will be taken with especial care. Multiplication will be measured after the addition of each new subassembly with the safety and control rods in, and after every three subassemblies also with the rods out. Close to criticality, the procedure of [fol. 4002] plotting the reciprocal of the multiplication factor against mass becomes especially accurate, and it will

be easy to predict exactly which subassembly will make the reactor critical.

The approach of a reactor to critical is no longer a hazardous procedure, but one which has been carried out dozens of times and has become routine. Moreover, the start-up of the PRDC reactor will be carried out by especially competent personnel who have been trained previously in this operation. Especially when criticality is near, the most competent members of the APDA staff and of its consultants, as well as experienced AEC representatives, will be present. I, therefore, believe that there is no hazard involved in the assembly of a critical mass in a populated area.

When the critical mass is finally reached, the control rods will be pulled out slowly, and the neutrons will be allowed to multiply until some low power level is attained, perhaps a few watts. At this low power level, the control rods will be calibrated, the safety rods tested, and the neutron responses to changes of reactivity measured.

At such low power level, the operation of the reactor can be completely predicted from theory because the fuel will not be heated, and there can be no effects from the bowing of fuel elements or from other mechanical features of the reactor. The behavior of the reactor then depends only on the properties of fission neutrons, in particular of the delayed neutrons, and is the same for fast and for thermal reactors, as I have pointed out before. The test at this low power level is, therefore, completely predictable and completely safe. Furthermore, the fuel is not radioactive at this time, because the reactor has not operated at high power levels.

If the reactor is found to behave satisfactorily at a power of a few watts, the power will then be raised to about 300 kilowatts of heat, which is only 0.1 per cent of the [fol. 4003] power at which the reactor is supposed to be normally operated. The coolant will be permitted to flow at its regular rate so that the temperature of the fuel will rise by only about 1/1000 of the amount it would rise at full power; i.e., a fraction of a degree. Such slight temperature changes cannot change the reactivity measurably, and in particular cannot lead to noticeable bowing. So the be-

havior should still be the same as at a power of a few watts, and this will be checked by experiments. The same will then be done at 1 per cent of the operational power level, where the same considerations still hold.

Then the power will be raised in slow steps. The steps presently proposed are 5, 12.5, 25, 50 and 100 per cent of the normal operating power, but this schedule is only tentative. At every stage a full set of experiments will be made to check the reactor behavior. The power will not be raised to the next level before a completely satisfactory result is obtained at the lower power level. As the power is raised, the fuel will expand and may possibly bow, and we shall thereby have a chance to examine the effect of fuel temperature on the reactivity and make sure that no positive temperature coefficient exists.

The chief instrument for studying the reactor behavior will be the reactor oscillator. This is a special control rod which can be moved rapidly in and out of the reactor. This control rod is made in the form of a cylinder, one half containing graphite, a good neutron moderator, and the other, containing boron-10, a strong neutron absorber, in the form of boron carbide. The rod can be rotated at any frequency from 0.01 to 25 cycles per second, and if it is so rotated, the reactivity will change regularly and periodically by a certain small amount. This amount can be selected by putting a suitable amount of boron-10 into the rod; it is planned to restrict the reactivity change to [fol. 4004] 6 cents. Then the rotation of this oscillator rod from the position where the boron is "out" of the reactor, to the "in" position, will cause such a small change of reactivity that no danger could result even if the rod stayed in the out position. Its 6 cents could easily be compensated many times over by any safety or control rod or by a rise of only about 30°F in the average reactor temperature.

When the oscillator is used to change the reactivity of the reactor, the reactor power will also oscillate with the same frequency. One can then measure the amplification, i.e., the ratio of the amplitude of the power oscillation to that of the reactivity oscillation. One can also measure whether or not the power is in phase with the reactivity, i.e., whether the maximum of power is reached at the same

time as the maximum reactivity, or if this is not the case, one can measure the difference in phase between the two oscillations. These two data, the phase lag and the amplification, will be measured with the oscillator oscillating at different frequencies, and ~~all this~~ information together will give a picture of the reactor characteristics.

I have developed the theory of these measurements in some detail and have published the results in a report, APDA-117, which is included as an exhibit to the PRDC License Application, as amended, and is incorporated by reference in this testimony. The theory shows that if the reactor has an instability (for instance, a resonance) at some high power, let us say at 300 megawatts of heat, one can predict this instability very well from experiments at much lower power, let us say at 50 to 100 megawatts. This is precisely the purpose of these experiments, namely to get advance warning of instabilities long before these instabilities actually occur, and this feature greatly contributes to the safety of oscillator experiments.

[fol. 4005] To make them still safer, we shall not simply increase the power by steps while keeping the coolant flow constant, but we shall first do experiments at lower power with reduced coolant flow. What we really want to study is the behavior of the reactor at different fuel temperatures because it is the temperature of the fuel which determines its expansion, its bowing, and its Doppler effect. Therefore, we should get essentially the same result if we operate at one-quarter of full power and one-quarter of full coolant flow as if we operated at full power and full coolant flow.

Staying at relatively low power and reduced coolant flow has several advantages. First of all, the total accumulated radioactivity will be kept low, so that if against all expectations there should be an accident, not many fission products would be released. Secondly, the slower the coolant flows, the slower also will be the responses of the reactor. We can then operate the oscillator at low frequency which is easier mechanically and again increases the safety.

To measure in particular the prompt positive temperature coefficient, we shall reduce the flow of coolant to a very small amount, possibly shutting it off completely, and we shall use very low power, perhaps 1 per cent of normal operating power. If then the power oscillation is in phase

with the reactivity oscillation, this indicates a positive prompt temperature coefficient; if it has the opposite phase, then the temperature coefficient is negative. If in this experiment, contrary to what I believe will be the fact, the temperature coefficient should turn out to be positive, we shall take the reactor apart and redesign it.

This method for measuring the prompt temperature coefficient is quite different from the transient one used in the EBR-I. Even if the prompt temperature coefficient is positive, our method will be completely safe. This is be-[fol. 4006] cause on the average the reactor fuel is not heated up. The power is alternately higher and lower than it was without the oscillator working, so that on the average it is not changed and the average fuel temperature and the average reactivity likewise remain unchanged. If there were any small change of average reactivity, this could be compensated by re-setting the main control rod. If there were any increase of average temperature, this could only be very slow and would then be communicated to the whole reactor, and, as I have mentioned before, the *over-all* temperature coefficient, corresponding to heating of the whole reactor, was negative even in EBR-I, and, therefore, this procedure is safe.

To measure the prompt temperature coefficient, we must change the temperature of the fuel alone, and we must do this very rapidly. This rapid change is provided in our method by rotating the reactor oscillator with high frequency. It is quite unnecessary to oscillate the reactivity with a large amplitude; a few cents is sufficient if we have sensitive instruments to observe the power oscillation. Even at the moment when the oscillator is completely "out," the excess reactivity will only be a few cents (which causes no hazard), and a moment later it is minus the same number of cents.

In the EBR-I transient experiment, on the other hand, the way by which the fuel temperature could be changed rapidly was to provide a large excess reactivity, in the end about 75 cents. This is relatively dangerous; the fuel heated rapidly without heating the rest of the reactor structure, and the reactor went out of control. We have no intention of using this hazardous procedure. By providing rapid

change without appreciable excess reactivity, our procedure will be safe and will still give the same information.

If, contrary to my expectations, our measurements do show a positive prompt temperature coefficient, we shall, of course, try to find its cause, which is likely to be some [fol. 4007] feature in the mechanical design of the fuel element. We shall then redesign the relevant part of the reactor core, put it together again, and repeat the tests. The reactor structure is sufficiently flexible to permit redesign without rebuilding the reactor vessel, the shield and other major parts.

A positive prompt temperature coefficient is not the only possible reactor instability; there is also the possibility of a resonance. As I have mentioned before, the EBR-I had such a resonance, at a frequency of about 0.4 cycles per second at full power and reduced coolant flow. Such a resonance is not in itself dangerous, but it makes it impossible to operate the reactor above the critical power at which it occurs. There is good hope that the resonance of EBR-I, together with the positive temperature coefficient, can be eliminated by a better mechanical design and that the mechanical design of the PRDC reactor has in fact accomplished this. The reactor oscillator is particularly suitable for finding a resonance, and if one exists, however, just as in the case of the positive temperature coefficient, it will be possible to predict the existence of a resonance at a much lower power than that at which the resonance occurs.

Summarizing, it is my opinion that safety tests with the help of a reactor oscillator can be carried out without danger to public safety and that the tests will give all the information required to insure safe operation of the reactor subsequently.

(10) *Safety Consciousness of PRDC*

As a consultant, I am not directly a member of the APDA staff, and thus I have an opportunity to observe their operation from the outside, though very closely. I am also in a position to make comparisons with other reactor laboratories with which I have been or am presently associated.

[fol. 4008] I am happy to be able to state that, in my experience, PRDC, and APDA, as well as the Dow-Detroit Edison group, have always been fully conscious of the requirements of safety. Because of their concern about safety, APDA arranged a meeting on the safety of fast reactors in Detroit in the Fall of 1954. This was the first meeting on this subject of such broad scope. Only one similar meeting had been held previously, under the auspices of the AEC, in New York City about 1949, but that meeting was much smaller and less comprehensive.

The safety meeting at Detroit was attended by many scientists from AEC and industrial laboratories and was considered very much worthwhile by all participants. The full record of the meeting has been published. It was the consensus of the participants in the meeting that the safety of fast reactors was not an extremely difficult problem and that most of the outstanding problems were on the way to being solved.

Further evidence of the concern of APDA for safety is shown by the fact that in my consultant's contract with this company, reactor safety and reactor design were specified as the two major problems with which I should be concerned.

Early in 1955, APDA urged that oscillator experiments be done on EBR-I, and these led to the discovery of a resonance in that reactor. Scientists from APDA and NDA, at the request of APDA, made major contributions to the early analysis of the resonance experiments. The fact that the resonance remained unexplained was the main reason ANL undertook the further experiments which finally led to the often mentioned accident to EBR-I. Full understanding is the primary requirement for safety, and even the lessons from the EBR-I accident itself are likely to help us to insure that no accident will occur in the PRDC reactor.

[fol. 4009] APDA initiated calculations at NDA of possible reactor accidents, considering the worst possible circumstances. At the request and expense of APDA, NDA made the first reliable calculations of the Doppler effect in fast reactors, which were then followed by calculations of the British Project and by some of my own.

Early in 1956, APDA initiated designs of the PRDC reactor, which would either avoid bowing or make the

temperature coefficient due to bowing negative. APDA had long been aware of the problem of the positive temperature coefficient of EBR-I and of the meltdown problem, and was engaged in programs for their solution. Since last summer, these programs have been intensified. There is close coordination between APDA and Argonne National Laboratory, both in fast reactor design and in safety programs.

(11) *Summary*

By the application of theoretical physics to what we now know, it is my opinion that a fast breeder reactor of the general type proposed by PRDC can be constructed and operated in a populated community without undue risk to the public, and that it can be demonstrated, when such reactor has been built, that its operation is safe.

The information to be developed by the projected program of APDA, including pre-operational testing, and by the related programs of ANL and other agencies of the AEC, will in my opinion eliminate any uncertainties that may now exist with respect to operation of the proposed PRDC reactor. I believe that this information will be sufficient to enable PRDC to furnish such additional technical information as is required to complete its Application for License under Section 104 b of the Atomic Energy Act of 1954.

On the basis of what we already know as a result of experimentation and research, operating experience with reactors, and theoretical calculations, it is my further [fol. 4010] opinion that such additional information will in fact show, with very high probability, that a reactor constructed in accordance with the present PRDC design can be safely operated. Even if the planned investigations should not demonstrate safety on the basis of the present design, they will at least substantially eliminate areas of uncertainty and point ~~the way to changes~~ in particular features of the design so as to assure safety. The design of the PRDC reactor is sufficiently flexible to accommodate such changes.

[fol. 4012] In evidence January 8, 1957, Tr. 43

BEFORE THE ATOMIC ENERGY COMMISSION

Docket No. F-16

In The Matter of

POWER REACTOR DEVELOPMENT COMPANY

TESTIMONY OF NORMAN HILBERRY,
DEPUTY DIRECTOR, ARGONNE NATIONAL LABORATORY

Background and Qualifications

My name is Norman Hilberry. I am the Deputy Director of Argonne National Laboratory. Since July 1, 1956 I have been acting in this capacity as the Laboratory's chief executive officer.

I was born in Lakewood, Ohio, now a suburb of Cleveland, on March 11, 1899. I received my Bachelor's degree with majors in both physics and mathematics from Oberlin College in 1921, started graduate work in physics at the University of Chicago in the fall of 1921, and accepted a position as instructor in the Physics Department at New York University in the fall of 1925. I taught both undergraduate and graduate physics courses at New York University, first at Washington Square College from 1925 to 1928 and then at the University Heights campus as an instructor from 1928 to 1930 and as an Assistant Professor from 1930 until the summer of 1940, at which time I returned to the University of Chicago on sabbatical leave and completed my graduate work. I received the Ph. D. in physics from the University of Chicago in the spring of 1941. The summer of 1941 was spent as a member of a State Department-sponsored cosmic ray expedition to South America. After a month utilized in making high altitude cosmic ray measurements in the Andes of southern Peru, I was a member of a symposium on cosmic rays sponsored by the Brazilian Academy of Sciences in Rio

de Janiero, returning to ~~New York~~ University in the fall of 1941.

[fol. 4013] In December of 1941 I was asked by Dr Arthur H. Compton to serve as his assistant on a war project which was just then being organized. This project was known during the war as the Metallurgical Project and since has been renamed the Plutonium Project. I reported for full-time duty on January 1, 1942 and have served in the national atomic energy activities continuously since that time.

During the early part of 1942 I had an active role in organizing the Metallurgical Project, not only by assisting in establishing it as an administrative unit but also by establishing and staffing new scientific divisions to meet the swiftly broadening demands for the incorporation of additional scientific and technical disciplines. During this same period my responsibilities included arranging for the necessary developmental work on methods for the production of the extremely pure graphite necessary for nuclear reactor construction and for obtaining this graphite in quantities adequate to meet the project needs. My duties also involved an active part in the procurement of the necessary high purity uranium metal. I was one of a group of three who served as the first site selection board for an atomic energy facility, recommending the selection of the present Oak Ridge site in the spring of 1942. Shortly after our selection of a site had been made, the Manhattan District of the United States Army Engineer Corps was organized, and our recommendation was forwarded to them for action, resulting in the acquisition of the present Oak Ridge site. As a member of the group present in the West Stands laboratory at the University of Chicago when the [fol. 4014] first nuclear reactor went critical on December 2, 1942, it was my assignment to man one of the several manual safety control devices provided for "emergency shut-down."

During the calendar year 1942, the activities of the Metallurgical Project expanded rapidly. In addition to the Metallurgical Laboratory itself, additional project groups became necessary and these were established under separate contracts at numerous other locations, such as Iowa State

College at Ames, Iowa, the University of California at Berkeley, California, Battelle Memorial Institute at Columbus, Ohio, and Washington University at St. Louis, Missouri. It was necessary to provide some organizational mechanism to handle all matters of technical administration for these associated contracts and to assure that the scientific program of the project as a whole was effectively coordinated. For this purpose, a Metallurgical Project Office, independent of the Metallurgical Laboratory, was established, with Dr. Arthur H. Compton as Director, to serve as the central executive unit for the Project as a whole. From January 1, 1942 until May, 1943, this Project Office reported on matters of both program and administration to the Office of Scientific Research and Development, which was the governmental agency responsible during this period for the over-all atomic energy program. From May, 1943 until July, 1946 it reported to the Manhattan District, U. S. Army Engineer Corps, and it was discontinued July 1, 1946. First as assistant to Dr. Compton and later as Associate Project Director, I served as the executive officer for this Project Office.

In the spring of 1942, active steps were taken to design [fol. 4015] a reactor of large enough power to produce experimental quantities of plutonium and a chemical process plant adequate to separate this plutonium from the uranium fuel elements in which it was formed. These facilities were designed with the assistance of E. I. duPont de Nemours & Co., Inc., and constructed at the Oak Ridge site. They formed the key facilities of the Clinton Laboratories which were established, staffed, and operated during the war by the University of Chicago. Construction of these facilities proceeded during 1943, and the X-10 reactor went critical in November of that year. As the representative of the Metallurgical Project Office, I was present at Clinton Laboratories during the final stages of the construction and throughout the start-up of this reactor.

During this period also, construction was completed in the Argonne Forest section of the Palos Park Division of the Cook County Forest Preserve of facilities to house the first nuclear reactor. The reactor itself was disassembled

from its location in the West Stands and reassembled in the new laboratory facilities. It was here also that the world's first heavy water moderated reactor was built and put into operation in May of 1944. Although the Argonne Forest laboratory was administered as a separate unit, it was operated under the same prime contract as the Metallurgical Laboratory itself.

In January, 1943, active design was started on a full-scale plutonium production plant. The Metallurgical Project was responsible for all matters of basic scientific and technical design, while E. I. duPont de Nemours & Co., Inc., was responsible for all detailed engineering design for construction and eventual operation of the plant. As a matter of contractual obligation, the Metallurgical Project was required to approval all final drawings, specifications, [fol. 4016] and initial operating procedures for the Hanford plant from the point of view of operability as far as basic science and technology were concerned.

In July, 1944, I was appointed Associate Project Director of the Metallurgical Project and assigned to Hanford as the liaison between the duPont company and all of the Metallurgical Project laboratories, in particular the Metallurgical Laboratory itself and the Clinton Laboratories. This role called for an active part in the start-up of the Hanford reactors in the fall of 1944.

During 1945 and the first half of 1946, the responsibilities of the Associate Project Director included the planning for re-establishment of the wartime atomic energy projects on a permanent peacetime basis and for the establishment of certain new atomic energy installations, including Brookhaven National Laboratory and the reorganization of the Metallurgical Laboratory and the Argonne Forest Laboratory into the present Argonne National Laboratory.

With the final establishment of Argonne National Laboratory on July 1, 1946, I became Associate Director, and on December 1, 1950, I was appointed Deputy Director. As a result, throughout the past ten years I have been in active contact with all of the Laboratory's reactor activities. These have included the first Experimental Breeder Reactor, EBR-I; certain phases of the design of the Materials Testing Reactor; the design, construction, and

operation of the Laboratory's heavy water research reactor, CP-5; the conceptual design and basic engineering of the Mark I Submarine Thermal Reactor; the conceptual design of the Savannah River Production Reactors; the design, construction, operation, and experimental programs associated with the BORAX reactors; the design and [fol. 4017] construction of the Experimental Boiling Water Reactor; the design of the Experimental Breeder Reactor, EBR-II; the design of the Argonne Low Power Reactor, to serve as a package reactor for use as a power supply in remote locations; the design and construction of the Argonaut, a training and limited research type of reactor; and the conceptual design of the ARBOR reactor, to meet full-scale engineering development requirements.

Incidental to these major reactor projects, my duties have required continual contact with the Laboratory's Zero Power Reactor program, which has included the zero power version of the Submarine Thermal Reactor, ZPR-I; the zero power mock-up of the Savannah River reactor, ZPR-II; the zero power mock-up of the Experimental Breeder Reactor, ZPR-III; the fast exponential experiments on fast reactors, ZPR-IV; and the coupled fast-thermal zero power reactor, ZPR-V. These projects have spanned the ten-year period since Argonne National Laboratory was established and have provided essentially continuous contact with highly important and productive reactor development activities.

The Laboratory is now preparing for the start-up and operation of the Experimental Boiling Water Reactor and of the Borax-IV reactor. Criticality experiments are under way with both reactors, and full-scale operation is scheduled in the immediate future. Argonne is also currently engaged in preparing detailed plans for rebuilding and operating EBR-I, and for the design, construction, start-up, and operation of a new Experimental Breeder Reactor, EBR-II. The Laboratory has completed the initial design of this [fol. 4018] new reactor, and is presently directing the activities of the architect-engineer on producing the detailed engineering design. The Argonaut reactor for university training and reactor research is nearing completion, and start-up is scheduled for the immediate future. The ARBOR reactor has been approved, and an architect-engineer is

being selected preparatory to proceeding with final design and construction. As the Laboratory's acting chief executive officer, I am administratively responsible for the execution of this program.

In all of these many reactor projects, commencing with the construction of the first nuclear reactor under the West Stands at the University of Chicago, considerations of safety and site location have been matters of primary concern to me and to others charged with responsibility for these projects. With other senior laboratory officers, I have taken an active part in the discussions upon which the decisions were based as far as most of these reactor projects have been concerned, and I have, of course, shared in the decisions as to site and safety features for those for which the Laboratory has had responsibility for construction and operation.

General Program

I am generally acquainted with the design of the proposed Power Reactor Development Company (PRDC) reactor and with the material included in its Application for License, as amended. I have also read the direct testimony to be given in this proceeding by Messrs. Amorosi, McCarthy, [fol. 4019] Bethe and Davis. I agree that the most important areas in which additional technical information must be supplied with respect to fast breeder reactors of the general type proposed are those which concern stability, temperature coefficients and effects of any credible melt-down. My testimony will be largely directed toward a brief summary of the information presently available in these areas, and an outline of the research program which is being carried out by Argonne National Laboratory in conjunction with other Atomic Energy Commission laboratories, Atomic Power Development Associates, Inc. (APDA), and others. In my opinion this program will not only provide much of the technical information in these areas which is needed to complete the PRDC License Application, but it will provide confirmation of the assurance which I believe we already have that a fast breeder reactor of the general type proposed can be so designed and constructed that it can be operated in a

populated area without undue risk to the health and safety of the public. There is, of course, no possible public hazard involved in any event in the mere *construction* of such a reactor, prior to the introduction into it of nuclear fuel.

The fast reactor program of Argonne National Laboratory is even older than the Laboratory itself; since before its establishment as a permanent laboratory in July, 1946, ANL and its predecessor have been actively engaged with the problem of fast breeder reactors. One of the final acts of the Manhattan District was to approve the Laboratory's proposal for studies leading to the design and eventual construction of a fast breeder reactor. These studies started in 1945 and have been carried on continuously to the present [fol. 4020] time. First as Associate Director and subsequently as Deputy Director of the Argonne National Laboratory, I was continually advised of and informed upon the planning and execution of this program.

These studies led to the design, construction, and operation of the first Experimental Breeder Reactor, EBR-I, which was completed in 1951. This was the first nuclear reactor to produce significant quantities of electrical power, providing sufficient power for self-contained operation of the plant and associated laboratory facilities. Initial operation was carried out on the Mark I core from 1951 to 1954. Measurements on this first core proved that even in this design the amount of plutonium produced was at least equal to the quantity of uranium-235 burned. Additional experiments were carried on with a Mark II core until November 29, 1955, at which time this core was rendered inoperative as the result of a fast excursion experiment, which I shall discuss further below.

Arising out of the studies of EBR-I, the Laboratory has undertaken a critical survey of the kinetic behavior of fast reactors, with the objective of determining more precisely the safety characteristics inherent in this class of reactor designs. These investigations will include studies of possible positive temperature coefficients and of possible operating instabilities which could arise as a result of temperature feed-backs. In addition to these kinetic studies which it is proposed to carry out on the present EBR-I reactor, the Laboratory, in cooperation with the Los Alamos Scientific

Laboratory, is also examining the possibilities of hazards which might arise in this type of reactor due to a postulated sudden compacting of the core arising from meltdown or [fol. 4021] other imagined nuclear incident.

In the meantime, in connection with the planning for a second experimental breeder of considerably larger size, a zero power reactor or critical experimental facility (ZPR-III) has been designed and constructed to carry out the necessary nuclear studies required in the design of a fast breeder reactor. This facility was placed in operation in October, 1955, and is presently engaged in a program of fast reactor studies.

Concurrently, design and development work needed to establish a definitive design for the new experimental breeder, EBR-II, has been carried out. As of the present time the first phase of the design work has been completed and an architect-engineer has been chosen to assist the Laboratory in producing the complete engineering design, including the necessary construction drawings.

In connection with design of the new Experimental Breeder, EBR-II, two ancillary programs are of significance. The one deals with the irradiation studies which the Laboratory is carrying on in connection with the determination of the irradiation stability of various fuel elements and fuel components and in particular, prototype designs for the EBR-II reactor and for the PRDC reactor. The other program deals with the continuing engineering studies associated with the technology of the use of sodium as a reactor coolant. This involves the engineering test of full-scale components, pumps, valves, seals, etc., using liquid sodium under the temperature conditions planned for the EBR-II reactor.

Since each of the above activities in the program of Argonne National Laboratory has a bearing on considerations of the inherent safety of the PRDC reactor, each will be presented in some detail.

[fol. 4022] *Experimental Breeder Reactor—EBR-I*

The Experimental Breeder Reactor, EBR-I, was designed initially to provide a facility to study the physics of fast

rea rs, particularly the breeding characteristics of such a reactor and secondly, to study the technology involved in the use of liquid metals as coolants for fast reactors. At the time these investigations were first started, fissionable material was scarce and urgently needed for military objectives. As a result, the reactor was designed for minimum critical size. This requirement resulted in an active core of very small size. The core volume was a hexagonal section essentially seven and one-half inches between flats and seven and one-half inches long. Since a major problem in fast reactor design is that of neutron leakage, it is clear that minor deviations in geometry in such a small core produce relatively much larger changes than would be produced by the same geometric deviation in a much larger core. The use of the small core also tends to make the heat transfer and temperature gradient problems more serious. Both of these problems would have dictated the use of a diluted core of larger volume which would simultaneously have increased the heat transfer area, reduced temperature gradients, and reduced the sensitivity to minor geometrical deviations. The diluted core, however, would have required larger quantities of fissionable material, and it was therefore not recommended. To determine the breeding inherent in this system, the enriched core was surrounded with a blanket of normal uranium. Radially this blanket consisted of an inner blanket of normal uranium rods cooled by the sodium-potassium [fol. 4023] alloy, and an external blanket outside the stainless steel vessel which contained the core, inner blanket and the coolant. This external blanket was composed of normal uranium "br cks" which were perforated with holes so that the "bricks" could be air-cooled. Longitudinal blanketing of the core was accomplished by placing normal uranium slugs in the stainless steel tubes, both above and below the highly enriched uranium-235 core slugs.

Since the EBR-I was to be an experimental unit, it was necessary that each fuel element be separately removable from the reactor. To accomplish this, the fuel elements were fabricated by inserting uranium slugs into properly fitted stainless steel tubes and bonding the uranium to the stainless steel with sodium-potassium alloy within the tube.

Such a fuel element then consisted of central uranium slugs of highly enriched uranium with normal uranium slugs inserted above and below to provide the upper and lower blankets. In the Mark I core, each stainless steel tube was provided with ribs which were milled on the outer surface. When the fuel element was inserted through the upper holding plate and into the lower holding plate and rotated, these ribs came almost in contact with the surfaces of the adjoining fuel elements. The sodium coolant then flowed through the apertures between these fuel elements.

The milling of these ribs on the steel tubes was an expensive and time-consuming operation, so that in the second core made for EBR-I, which we called the Mark II core, the ribs were eliminated and in consequence, the clearance between fuel elements was increased by about 50 mils. Two types of geometrical distortion are obviously possible within this core. One would be a bowing of the [fol. 4024] uranium slugs themselves within the stainless steel tubes. The thickness of the liquid bond between the uranium slugs and the stainless steel wall was 10 mils, so that a bowing of 20 mils within the stainless steel jacket would be possible. Since the flux across the reactor is not uniform but drops off from the center toward the edge, it is clear that the side of each slug toward the center is somewhat hotter than the side away from the center, and consequently, bowing can occur due to temperature differences across the slug. With the coolant flowing over the surface of the steel jackets, these temperature differences are clearly minimized. If the flow of coolant is stopped and in addition, the power is increased rapidly, these temperature differences are magnified, and bowing effects become much more probable.

The second source of geometrical deviation would be in the actual bowing of the stainless steel tubes themselves. In the Mark I core, the presence of the ribs reduced the possibility of this bowing to a matter of a few mils at most throughout the entire length of the tubes. In the Mark II core, however, bowing due to irregular heating could have amounted to as much as 50 mils. It should be noted that the temperature differentials are such that any bowing arising from either source would tend to move

the fuel slugs closer together at the center of the reactor and thus tend to increase the reactivity of the reactor as a whole.

Construction of EBR-I was completed in August, 1951, and the reactor placed in operation. During the lifetime of the Mark I core, the major emphasis was on steady operation to obtain breeding information as quickly as [fol. 4025] possible, and while some experiments were done, no extensive experimental program was carried out. The emphasis was on building up total megawatt days of operation and upon study of the engineering behavior of the system, including the electrical power production. Under certain start-up conditions, particularly when doing foil irradiations, certain operational anomalies were observed. These occurred only under extreme operating conditions and were totally absent during routine operation.

With the introduction of the second, or Mark II core, rather extensive experimental programs were undertaken to determine the nuclear characteristics of the reactor. During this phase of the reactor's operation, less emphasis was placed on routine operation and consequently, more occasions arose on which the operational anomalies were observed. These consisted of oscillations in power level when the flow rate was drastically reduced, and the appearance of a positive temperature coefficient under circumstances of start-up with very short flux doubling periods and with zero or no coolant flow. It should be emphasized that these anomalies were not characteristic of normal operation and required rather drastic abnormalities in operating procedures for their observation. Under normal operating procedures, the reactor had a definite negative temperature coefficient and very stable operating characteristics. Four years of essentially continuous trouble-free operation were experienced. In my opinion, this attests to its inherent reliability.

When the experimental program that had been outlined for the reactor was completed, the question immediately arose as to whether a final series of experiments should [fol. 4026] not be performed in an attempt to explain the anomalies which had been observed under abnormal

operating conditions. These conditions of reduced or of zero coolant flow and of start-up at very short periods are obviously only procedures undertaken for strictly experimental purposes. To carry out the experiments requires that two automatic safety controls—the flow interlock and the period meter interlock—both be overridden. It is clear that such experiments are fraught with possibilities of damage to the reactor. New high speed instrumentation was added to the reactor, the flow and period meter interlocks were disconnected, and the final series of experiments undertaken. A reactor oscillator was installed which made it possible to increase and decrease the reactor power at variable frequency under the conditions of reduced flow where instabilities earlier had been observed. It was found that for a given frequency of the oscillator, a "resonance" in the power output could be obtained. While harmonics of this resonance frequency were searched for, there was no clear indication of their existence, although if they were of relatively minor magnitude they might have been present without being observed.

The program then shifted to an [illegible]vation of the positive temperature coefficient. The coolant flow was stopped and the reactor started up on a fast enough period so that temperature differentials would be established in the fuel slugs. The characteristic of a positive temperature coefficient immediately appeared; that is, there was an increase in the rate at which the power doubled as the power itself increased. On the final experiment, in which [fol. 4027] the doubling period had been reduced to approximately 0.3 seconds, a delay in scramming the pile permitted the temperature to overshoot so that the uranium became heated above the temperature, roughly 725°C, at which the uranium-iron eutectic forms and the center of the core melted, forming the eutectic. Although the reactor was scrammed manually, the power limitation circuits which had been above the normal trip-level also operated, and it was only the fact that the reactor was on such a short doubling period that permitted it to overshoot to the extent that melting took place. From the control room, it was impossible to tell that anything had gone amiss, although

one thermocouple did indicate a temperature of 590° C, and since it was clear that this might not be a peak temperature, there was some suspicion that the actual temperature rise might have been appreciably higher than this. Within some 15 minutes, however, there were indications of abnormal fission activity both in the coolant and in the stack gas from the building. This activity could only have come from the rupture of a fuel element jacket. The amounts of activity were very low, and although the building was evacuated until precision measurements could be made, the occupants returned immediately thereafter.

Attempts to remove the fuel elements from the reactor core showed very promptly that melting had occurred. While some of the external rods could be removed, by far the greater number could not. This meant that the core would have to be removed as a complete unit from the reactor. The original design had provided for this operation, but it was necessary to erect on top of the reactor a temporary cave to provide radiation protection for the [fol. 4028] operators.

The core was allowed to cool from the time of the incident in November until June before an attempt was made to remove it from the reactor. In the meantime the temporary cave was erected and necessary remote manipulation devices provided. The core itself was removed without incident, placed in a specially designed coffin for radiation protection, and shipped to Argonne National Laboratory at Lemont for detailed examination in July, 1956. One of the caves at Argonne was made gas-tight so that the inspection of the core could be carried out in an inert atmosphere to reduce any possibility of the sodium which still clung to the fuel rods catching on fire and thus initiating a uranium fire. When the cave had been adapted, the core was removed from its coffin and examined in detail. I personally participated in this examination, and I also have studied the photographs and analyses made of the core. Most of the outside rods could be removed without difficulty. Many of the steel jackets of these outside rods, however, had been penetrated by iron-uranium eutectic which dissolves steel with great readiness. The iron-uranium eutectic formed in the center of the core had

obviously been ejected from the core center outwards between the fuel elements displacing sodium and interacting with the steel jackets of the outside fuel elements. The uranium slugs in the outside fuel elements were intact and undamaged.

In the center of the core, where the melting would first have occurred, no semblance of fuel tubes or fuel slugs could be observed. The average density of the core assembly before the incident was approximately 10.5 grams per [fol. 4029] cubic centimeter. This represented an average of the actual quantities of uranium, stainless steel, and sodium. The density of the material removed from the center of the core after the meltdown was approximately 4 grams per cubic centimeter. This means that much of the uranium originally in the center of the core after alloying with the iron had been ejected possibly by bubbles of boiling sodium-potassium alloy to the outer regions of the core. This central material was very porous, much like volcanic tuffa. Surrounding this central zone was a layer of slightly more dense material, with a density of approximately 6 or 7 grams per cubic centimeter, and outside of this zone; melting apparently did not occur. Directly below the center low-density core was a region in which there was material of approximately 14 or 15 grams per cubic centimeter, as if part of the eutectic had settled into this area and then solidified. Below this layer there had been no melting of the fuel elements and they could be readily broken away.

It is impossible to state that this melting phenomenon actually stopped the reactor in this instance, since the operator did hit the scram button and since the power limitation circuits also operated properly. The observed meltdown phenomenon therefore conceivably occurred after the reactor was shut down by its safety circuits. However, it is clear to me that the observed meltdown would have stopped the reactor if the safety circuits had not done so, although further melting might have occurred had the reactor had to wait for this mechanism to be fully operative. In my opinion, the results of this experiment indicate that in the process of a meltdown of a sodium-cooled fast reactor [fol. 4030] core the core tends to disassemble itself, reducing

the reactivity and power level and acting as an added and automatic safety device.

A more detailed analysis of the core after the meltdown phenomenon is now in progress, and this investigation will undoubtedly give further quantitative data on the effect of a meltdown in reducing the reactivity of a fast reactor core. In my opinion it will give an even more definitive indication that this phenomenon is an effective control measure in preventing major hazards in fast reactors of this general type.

This final series of experiments determined without question that the EBR-I Mark II core did have an unexplained instability at low coolant flow rates, and that it did have a positive temperature coefficient for zero flow and for fast start-up periods. The most probable explanation of this positive temperature coefficient is that it resulted from bowing, but the experiment itself, of course, offers no proof that this is true. The observed instabilities in turn may be due to some type of thermal feed-back, but again it is not clear from the experiment itself what the origin of this feed-back is. Consequently, it is proposed to carry out a new set of experiments on EBR-I which I believe will definitively answer these questions.

Two changes are being made. The basic coolant flow of the reactor is being modified. In both the Mark I and Mark II cores, the sodium-potassium alloy first flowed down through the inner blanket and then up through the core, from there out to the heat exchanger, to the pumps, to the hold-up gravity tank, and thence back to the inner blanket. This flow pattern is being modified so that the flow can either follow this series pattern or, as an alternative [fol. 4031] pattern, the sodium-potassium alloy can flow independently in parallel streams through the core and through the inner blanket simultaneously. (Parallel flow is presently planned for the proposed PRDC reactor.) This change from one flow pattern to the other should give an indication as to the nature of thermal feed-backs if they are associated with coolant flow when the oscillator experiment is repeated with the new core.

The second major change is that the third, or Mark III, core is being built as a fully rigid core. Instead of using

stainless tubes with uranium slugs thermally bonded to the stainless steel with sodium-potassium alloy, the new core is being formed of co-extruded zirconium jackets on the uranium. This will prevent any possibility of bowing of the fuel slugs within the tubes. Each fuel element is being provided with detachable ribs so that when the core is assembled and clamped together, it will form a rigid assembly in which bowing is practically impossible. If bowing is the source of the positive temperature coefficient, the design of the new core should reduce this positive temperature coefficient very greatly, if not completely eliminate it. As a check on this experiment, the core will be disassembled and the ribs will be removed except at the top and the bottom of each fuel assembly. When reclamped, the modified Mark III core will be rigid at the top and at the bottom but will be free to bow in the center. This will lead to an increased positive temperature coefficient in the modified case, if in fact the positive temperature coefficient is, as I believe it is, due to bowing. In any event, these experiments will provide an unmistakable check on whether this is the source of the positive temperature coefficient observed in EBR-I.

[fol. 4032] In my opinion, the proposed EBR-I program using the Mark III core will demonstrate that the positive temperature coefficient and probably the observed instability at low coolant flow rates were due to bowing of the fuel elements and to series coolant flow, or a combination of these factors. If this is so, these undesirable characteristics can be eliminated by relatively simple design changes, and will in fact be eliminated by the design presently proposed for the PRDC reactor as described in the PRDC Application for License. In any event, however, I believe that the proposed EBR-I program will almost certainly provide the answer to the source of the positive temperature coefficient and instabilities observed in EBR-I with the Mark II core, and when this answer is definitively obtained I am confident that these undesirable features can be eliminated by engineering or design changes.

Another important result which I believe will be obtained from the EBR-I program is verification of the validity of the use of an oscillator as a safe and reliable means for

exploring the operating characteristics of a fast reactor and the determination of the presence or absence of any operating instabilities. In my opinion, information already available indicates that this technique is both safe and reliable, but the oscillator experiments we plan to conduct on EBR-I should eliminate all reasonable doubt on this score.

The results of the EBR-I program should begin to be available during the summer of 1957. While it may prove profitable to carry on oscillator experiments as a continuing investigative tool for a relatively long period of time, it appears that the basic results for both investigations should [fol. 4033] be available within the coming year, which is in ample time for the application of any new data obtained to the final core designs of both the EBR-II and PRDC reactors.

Joint ANL-LASL Fast Reactor Safety Program

Quite apart from the further EBR-I program I have just described, the very fact that EBR-I gave four years of trouble-free operation using first the Mark I and then the Mark II core, and the further fact that the Mark II core was carried through the much-discussed and much worried-about "meltdown" with no damage except to the core itself, provides encouraging assurance with respect to the normal, power-production operation of a well-designed fast breeder reactor. In addition, however, a further extensive research and experimental program is planned with respect to the possible results of unpredicted concatenations of malfunctions, or of highly skilled sabotage of such a reactor.

Control of a reactor depends on the rate at which the reactor power increases and this in turn depends on how fast the reactor's neutron population increases. The population increase in turn depends on the percentage in numbers—the "excess reactivity"—per generation and on the length of time until each member of one generation has provided his progeny for the next. It happens that most of the neutrons provide their new progeny promptly, but there is always one group which dallies in making its

appearance, the "delayed" neutrons. These come not from the fission process itself but from nuclear reorganization within the fission product atoms produced by the fission act. [fol. 4034] Reorganizations take time, so that as long as the neutrons from this group are an essential part of those required to maintain or increase the population, it is their generation lifetime that determines the rate of increase.

Only when the excess reactivity is so great—"prompt critical"—that the prompt individuals themselves can maintain or increase—"excess above prompt critical"—the population does their characteristic generation time become controlling. One of the safety objectives in the design of any reactor—thermal or fast—is to avoid the possibility that the reactor may become "prompt critical." Since the time for "prompt" multiplication is much shorter than that for the "delayed" group, it is thus always the objective to design for a maximum excess reactivity obtainable in any reactor less than that required for prompt critical; if the total amount of excess reactivity available is not enough to make the reactor prompt critical, the danger of this occurring is obviously minimized. Due to the fact that absorption probabilities are in general higher for neutrons of velocities characteristic of thermal reactors than for those found in fast reactors and due particularly to the spectacular thermal absorption due to xenon, one of the products of the fission product decay process, it is much easier to meet this criterion of small excess reactivity in the design of a fast power producing reactor than it is for a similar purpose thermal reactor; in other words, a thermal reactor usually requires that considerably greater excess reactivity be available than does a fast reactor. This is one of the advantages of a fast reactor over a thermal reactor, and as a result the fast power reactor in normal operation is [fol. 4035] at least as safe as and in some ways safer than a thermal reactor designed for the same service.

On the other hand, if prompt critical is passed, the power in a fast reactor will increase a great many times faster than will be the case in a thermal reactor. The short "prompt" neutron lifetime associated with a fast reactor is the basis of the worry concerning what might happen if in fact it should somehow become prompt critical. Thus,

while a fast reactor can be so designed that untoward incidents are much less likely to happen than in the case of a thermal reactor, if such incidents do occur the results may theoretically be of much greater magnitude.

It should be kept clearly in mind that unless the magnitude of the accident is such as to breach the structure housing the reactor, the containment provided by it will prevent any damage beyond that to the reactor itself. It is conceivable that a structure could be designed strong enough to contain almost any specified explosion. The provision of an extreme housing facility would obviously be a last resort in any power plant designed to be economically feasible, however, because of its excessive cost. This makes it imperative to determine the actual situation with regard to such incidents as completely as is possible. In the case of the EBR-I incident, the results were insignificant. While the core was destroyed, the remainder of the reactor, including the inner blanket, was undamaged. Other available information indicates that under no circumstances developable in a fast reactor could an explosion occur which would even approach that of an A-bomb. It is clear that there [fol. 4036] is no credible possibility that a reactor could be a bomb. The problem on which we are still working, however, is to determine just what forces could be released by the maximum accident which might credibly occur.

A joint program between Los Alamos Scientific Laboratory and Argonne National Laboratory has been established to obtain the answers. Los Alamos has well tested computational techniques for determining the explosive efficiency of various assembly geometries and assembly procedures and has the needed experimental data to interpret the calculations. Argonne has a major backlog of calculations and experimental results on fast reactor designs, including reactor control and kinetics. It is believed that the joint effort of the two groups, each bringing its own specialized competence to bear on this problem, will provide the information needed to give positive answers at least within the "order of magnitude" range within the year. I believe that the results of this work will show that the maximum credible accident of a reactor of the general type proposed by PRDC will be containable.

In order to make assurance doubly sure, Argonne National Laboratory is also proposing to build a nuclear "meltdown" facility to determine precisely the mechanics of the meltdown phenomenon for specific types of fuel elements and for sub-assemblies of such fuel elements. This will be done with the test specimens both in air and in static liquid metal coolant. It is believed that a significant explosion can occur only if the core is first melted. If this be true, these experiments on the mechanism of melting become important, as they will then supply the detailed information needed for precise calculations of the type [fol. 4037] outlined above. The Los Alamos group have agreed to assist with the really difficult instrumentation problems inherent in obtaining quantitative data from such "fast excursion" experiments. While no guarantee of success can be given, it is hoped to have such a facility ready for operation within a year to eighteen months in order to provide data to the group carrying out the explosion calculations at the earliest date possible. With such data, the answers to the "containment" problem should be essentially definitive and they should be available by January, 1959, in advance of both the EBR-II and the PRDC reactor start-ups.

It is not, of course, absolutely certain at this time that such a nuclear meltdown facility can be satisfactorily designed and built in time to meet this schedule, but the work is being vigorously pursued and it is hoped that we will be able to do so. While such actual experimentation is not in my opinion essential to a final safety determination, such experimentation would provide additional verification of the computations being made by furnishing an experimental check on the nature of the "assembly procedures" assumed.

Fast Zero Power Reactor Program (ZPR-III)

The art of reactor computation has for several years been essentially an exact science for uranium fueled thermal reactors. Until very recently, however, this was not true for fast reactors. As late as 1951 even the critical mass calculation for EBR-I was in error by more than twenty-five per cent. It was clear that energetic steps had to be

taken to remedy this situation. Detailed study of all of [fol. 4038] the many micro cross-sections involved and their variations with energy throughout the energy range from fission energy down to thermal energy would have been the direct and straightforward approach. Unfortunately this approach would not have provided the data in time. A more direct, if less elegant, approach was essential for the prompt development of fast reactor systems.

The first approach taken was to utilize the EBR-I itself as an experimental tool to determine as many of the parameters needed for fast reactor calculations as possible. In this connection a major effort was made to determine the neutron "energy spectrum" throughout the reactor core. This involved the development of new measuring devices and new experimental techniques. The study did give a measure of the relative numbers of neutrons as a function of velocity from fission energy down to thermal at various points throughout the core and blanket. This made it possible to check the validity of "educated guesses" as to the values of various numbers used in the fast reactor calculations. It was also possible to measure the "averaged" cross-sections of important materials in a variety of quite different neutron velocity distributions. This again provided valuable new data which could be used to check computational methods.

It was clear that EBR-I itself was too inflexible to provide the wide variety of data needed to explore all the facets of fast reactor theory. It was fine for the objectives for which it had been designed but the very fact that it was primarily a power reactor meant that it lacked the flexibility [fol. 4039] imperative in an experimental facility for a thorough and complete investigation of fast reactor physics. Consequently, the Laboratory proposed a Fast Zero Power Reactor, ZPR-III, which was designed specifically to serve this purpose. Moreover, it would serve at a later date as a facility for checking the detailed reactor physics of the EBR-II core and of the PRDC core long before those reactors were ready for start-up. The Atomic Energy Commission approved the proposal, the facility was designed, and in order to provide the greatest possible flexibility in experimental program, it was built at

Argonne's Idaho Division laboratory at the National Reactor Testing Station in Idaho. It was placed in operation in October of 1955 and is now engaged in a program of experiments to provide the data required for the improvement of the precision and reliability of fast reactor calculations. Agreement between calculation and experimental observation has already become excellent and has served to verify many quantitative predictions of fast reactor theory. As this program continues, I am sure that it will serve to make the art of fast reactor computation as exact a science as thermal reactor computation has become.

It was clear when approval was given for ZPR-III that it would take many months for its completion. Consequently, ZPR-I was rebuilt at the Chicago site to provide an essentially zero power thermal "source pile" to serve as a copious neutron supply for a fast "exponential" or "sub-critical" assembly. This unit, ZPR-IV, made possible an early start on the problem of determining the parameters needed for fast reactor calculations. Since ZPR-III, being self-critical, is so much more sensitive and potent a tool for [fol. 4040] the purpose than ZPR-IV, the latter was decommissioned shortly after the ZPR-III start-up. It had served its purpose in pushing ahead the schedule for establishing a sound basis for fast reactor theory.

Not only have ZPR-IV and ZPR-III provided the basis on which fast reactor calculations can be carried out with full confidence in their results, but ZPR-III has provided another real contribution to confidence in inherent fast reactor safety. One haunting area of ignorance in reactor theory in general has been the question of the so-called "Doppler effect." One way that nuclei capture neutrons in their immediate neighborhood is through "resonance." Put very crudely, if a neutron is travelling at just the right speed the nucleus acts as if it had a revolving door which let the neutron pass right inside. If the neutron is going either faster or slower than this "resonance speed" it acts as if it bounced back or got knocked out of the way. There is a little latitude in the speed it can have, but not much.

Obviously this mechanism strictly limits the number of neutrons that can enter this nucleus. If this particular

"resonance" lets in neutrons which result in fission, it is clear that this limits the number of fissions that are going to be produced; the nuclei have to wait around until a neutron of just the right speed comes along. Now if you "heat up" the nuclei, this merely means that you start them dashing around with speeds of their own, the speed depending on how hot they get, the hotter, the faster. Now, if a nucleus dashes toward a neutron, the neutron itself will have to be going a little slower than the "resonance speed" if it wants to get through the revolving door. It [fol. 4041] would have to be going faster than the "resonance speed" to get in the door if the nucleus were running away from it. It is clear, however, that with both neutrons and nuclei in motion there is an increased number of neutrons—those with somewhat higher or lower speeds than the "resonance speed"—that can now enter and produce fission. The "resonance" has been "broadened." This would mean that if the temperature of the fuel in a reactor were raised it would result in a greater number of fissions than when cool, and other things being equal the number of progeny in succeeding generations would go up—the reactor would have a "positive temperature coefficient." Clearly the effect exists, but how big is it? Could the "positive" coefficient of EBR-I be due to a Doppler broadening? If the effect were of the magnitude which reasonable calculations would predict, the answer was "no" but there were no experimental data to prove the validity of the calculation. There might be an anomalous effect of which we were ignorant.

To test this experimentally the EBR-I core was mocked up in ZPR-III in such a way that fuel could be electrically heated and actual measurements made on the change in reactivity. The experiment is extremely difficult to perform because the effects expected are small. However, it was shown in the very first experiments that the effect is no larger than the theoretical calculation predicted within the temperature range it was possible to use in the experiment. Further work using more sensitive instrumentation and a larger temperature range has been completed and provides data accurate enough to make possible reliable calculations. The observed effect was only one-fifth of that pre-

[fol. 4042] dicted by early calculations. It is quite clear that this particular effect is not responsible for the positive coefficient in EBR-I. It will provide no problem in either the EBR-II or the PRDC reactors. This answer is now definite.

In addition to the general fast reactor physics experiments of the kinds so far described, ZPR-III is scheduled to make detailed studies of the physics of both EBR-II and the PRDC reactor. These investigations will include critical size determinations and detailed flux distribution studies. The latter will provide the power distribution data required for final design check on coolant flow, core stress, etc., before the reactor goes into operation. The control rod patterns will be studied in detail both as to "reactivity worth" and as to modification of flux and therefore power distribution. Danger coefficient measurements will be made to determine the relative importance from a reactivity standpoint of the different fuel locations within the core. Finally, the effect on the reactivity of displacing part of the fuel axially out of the core to one end, simulating the effect of melting and dripping, will be determined in order to give a measure of the way in which the reactivity would change and how rapidly if part of the core were to melt. The PRDC reactor experiments will be undertaken beginning early in 1958, depending on how fast the necessary fuel pieces can be fabricated. The complete set of experiments will require approximately a year for completion. In the meantime the EBR-II experiments will be carried out. Since it is not clear that they can be completed by the end of 1957 and since all of the important general experiments [fol. 4043] would be delayed for two years under the above program, the Atomic Energy Commission has authorized the construction of still an additional Zero Power Reactor facility. The Laboratory is presently engaged in the design of this facility and hopes to have it in operation early in 1958. This will make possible full investigation of the problems of both EBR-II and of the PRDC reactor without interference between them. In both cases the investigations will be completed in adequate time to meet the established schedules for the respective reactors.

It will be noted from this discussion that months of experimental work must be done before definitive answers are available to be supplied to the Commission as a basis for its determination as to the ultimate safety of any particular fast breeder reactor. Some of the necessary studies have been under way more than a year. The construction schedules of EBR-II and the PRDC reactor are such that the experimental data will be available in time to be utilized.

The Second Experimental Breeder (EBR-II) Program

The first experimental breeder reactor, EBR-I, was designed to meet strictly limited and highly specific objectives. The first of these was to prove that breeding—the production of more fissionable material during reactor operation than the reactor burns—was not only a theoretical possibility but an actual, technically achievable goal for future power reactor designs. The second objective was to establish the feasibility of the use of a liquid metal coolant—sodium potassium alloy in this case—as a sound [fol. 4044] engineering technology which could be relied upon in power reactor design work. The third objective was to demonstrate that even such an experimental device could produce significant quantities of reliable electrical power.

EBR-I fulfilled its objectives with signal success. It demonstrated that in its operation at least as much plutonium was produced as there was uranium-235 fuel burned. The sodium potassium alloy cooling system yielded trouble-free operation throughout the four years that the reactor was in steady operation after initial shakedown difficulties were remedied. Finally, upon demand, the electrical generating system provided the two hundred or so kilowatts of electrical energy required to operate the reactor and the associated laboratory activities in a thoroughly reliable manner.

The importance of this proof of the technical feasibility of breeding should be clearly realized. "Coal" which consisted of one hundred thirty-nine parts of rock and one part of burnable carbon would be considered pretty poor stuff as a fuel. Conceivably, the one part of carbon might be separated out and burned, but clearly the use of coal would

be quite a different matter were this to be the actual situation. This is, however, almost the situation where normal uranium is concerned. Normal uranium contains only one part of the fissionable uranium-235 to one hundred thirty-nine parts of uranium-238. The uranium-235 can be separated and burned, but the process is costly and it leaves the uranium-238 as a practically useless residue.

Imagine now that by some strange scientific process, when the one part of carbon is burned within the one hundred [fol. 4045] thirty-nine parts of rock it transforms a part of this rock into a new burnable material and this new material in burning in its turn transforms still more rock to new burnable material. If for every pound of carbon or new burnable material burned a pound of new burnable material were formed, it is clear that most of the "coal" would be burned; thus, although it might look like poor stuff at first glance, in fact it would make a usable fuel without the costly separation process. Moreover, it now becomes an economic fuel source because it represents a total energy supply one hundred and forty times as large as that available in the original carbon. Clearly, the separation of the carbon to be burned separately, thus rendering the rock useless, could only be justified by high priority needs. The world's fuel supplies are finite, and willfully to dissipate such a major fraction of one of the potentially major energy sources would be inconceivable.

No such scientific process is known for our postulated carbon-rock composite "coal"; however, such a process *is* known for the uranium-235–uranium-238 fuel.

In the act of fissioning the uranium-235 atom bursts into two fragments, thus forming two smaller atoms, and at the same time it ejects some left-over neutrons. Ideally, a fission occurring in pure uranium-235 would only need to eject one left-over neutron in order to produce a continuing reaction, provided the body of uranium-235 were infinite in size. Each left-over neutron would eventually enter a uranium-235 nucleus, produce the fission act, and result in a new left-over neutron. This would simply keep on going [fol. 4046] indefinitely. The ideal case is not quite as simple as pictured, however, because occasionally a left-over neutron enters a uranium-235 nucleus in such a way that it can

be tolerated without causing the fission eruption; it produces uranium-236 instead. This being the case, each fission act must produce on the average more than a single left-over neutron if the reaction is to continue.

In reality the situation is much worse. No body of material is infinite in size so some of the left-over neutrons leak out of the uranium-235. Others are absorbed in the structural materials used to position the uranium. Since the probability that a neutron will cause fission in a uranium-235 atom increases as the neutron speed is reduced, moderating materials are introduced into the system so that the neutrons will get slowed down like a sprinter dashing into a mob of holiday shoppers—he is soon slowed to the pace of the crowd. These moderators also absorb some of the left-over neutrons. Moreover, each fission act produces heat. To remove the heat requires a cooling system and this in turn provides new materials to absorb the left-over neutrons. Fortunately, when uranium-235 fissions, it produces on the average approximately 2.5 left-over neutrons per fission act. This number is large enough so that, with care, every fission act leaves at least one left-over neutron to carry on the reaction, the others being absorbed in the materials of construction or controls, or leaking out of the system.

When one of the materials of construction happens to be uranium-238 a new phenomenon enters the picture. This material, like other materials of construction, absorbs neutrons, but in this instance the neutron transforms the [fol. 4047] uranium-238 into uranium-239, which by radioactive decay becomes plutonium-239. Plutonium-239 is a "burnable" material in the same sense that uranium-235 is "burnable." If now by careful design the loss of left-over neutrons to leakage, to absorption, and to structural materials and controls can be kept to less than half of a left-over neutron for each fission that occurs, one neutron would be left for capture in uranium-238 to produce plutonium, and the quantity of plutonium so produced would in this instance be equal to the quantity of uranium-235 burned.

In a reactor burning pure uranium-235, fission-produced ash builds up as the burning process continues, and these

fission products are in many instances strong neutron absorbers, particularly for neutrons that have been slowed down to thermal energies. This means that in effect, new fuel must be continually added to the reactor to maintain the general reaction, otherwise the absorption effects would soon rob the necessary left-over neutron from the uranium-235 reaction. Rather than add new fuel continuously, it is customary to put in originally more fuel than is needed and to balance this excess fuel with excess absorption in the form of control rods which can be removed as fission-produced poisons build up.

In a reactor using uranium-238 as well as uranium-235, the formation of new burnable material in the form of plutonium-239 makes necessary a much smaller allowance in excess fuel put in originally to take care of burn-up. The uranium-238 transformed to plutonium-239 is the equivalent of the addition of fuel or the removal of a control rod. Theoretically, it should be possible to design a re-[fol. 4048] actor in which the neutron economy is so good that the production of plutonium and its subsequent burning to produce additional plutonium would enable the reactor to burn practically all of the uranium-238. Practically, this is impossible in a heterogeneous reactor because the accumulation of fission products eventually causes changes in the natural character of the fuel element, such that the fuel element must be removed for reprocessing in order to remove the fission products and restore it to its original metallurgical condition. Nevertheless, the production or breeding of plutonium from the burning of uranium-235 or other nuclear fuel is of enormous importance from the standpoint of fuel conservation as well as reactor economy and efficiency.

Thermal or moderated reactors were the first to be built because the probability of a slow neutron being captured in uranium-235 is very much higher than it is if the same neutron had essentially fission energy. This means that a much smaller quantity of uranium-235 is required to reach critical size, with consequent reduction of fuel inventory.

On the other hand, the absorption coefficients of various structural materials and the probability that uranium-235 or, more importantly, that plutonium-239 will absorb a neu-

tron without producing fission are very much less for neutrons of high energy—that is, moving with extremely high speeds—than for thermal energies. This decrease of capture probability also holds for all of the fission product ash that is produced in the fissioning of uranium-235 and plutonium-239. Moreover, uranium-238 itself occasionally will fission when it absorbs an extremely fast neutron. These factors make it clear that it should be possible to ob- [fol. 4049] tain a very much better neutron economy in a fast reactor than is possible with a thermal reactor. Added to the above considerations, if the neutrons are to be kept at as high energies as possible, no moderator is needed and consequently losses by absorption to a moderator are also eliminated. The price that one pays for this increased neutron economy lies in the fact that the probability of fission has decreased and this in turn means a larger critical mass and therefore greater fuel inventory.

Here again it should be pointed out that in the fast breeder, due to the formation of new fissionable material in quantities equal to or greater than the quantity burned and due to the lowered probabilities of absorption, the amount of excess reactivity which must be built into the reactor to provide satisfactory operating periods between reloadings is very small compared with the thermal reactor. In the PRDC reactor, for example, the amount of excess reactivity that must be built into the reactor and balanced by the control at operating temperature is less than that required to take the reactor to prompt critical.

To summarize, if uranium is to supply the significant energy reserves for the future which it should supply, uranium-235 "burner-uppers" are definitely ruled out. Nuclear reactors for power purposes must be designed with such high neutron economy that at the worst, approximately one new atom of fissionable material is produced for each atom fissioned, or, if possible, even more fissionable material is produced than is burned. The clearest immediate road to this goal lies in the design of fast-neutron breeder reactors, and this type of reactor presently gives the only [fol. 4050] technologically proven avenue toward achieving this goal.

Against this background the EBR-I accomplishments become highly significant. As pointed out earlier, EBR-I was purposely made as small as possible to cut down on the required quantities of uranium-235. This very smallness insured high leakage. If the EBR-I would breed at all, a large power breeder could be designed to produce more fissionable material than it burned with a high degree of certainty. That the EBR-I did produce at least as much fissionable material as it used assured the nation that nuclear power is not only a specialty power source dependent on pure separated uranium-235, but also a potential economically competitive central station power source with essentially all of the uranium—both 235 and 238—available for energy production. Further, EBR-I demonstrated that liquid metal cooling provided not only a technically feasible system but also one compatible with a successful breeding neutron economy. The problems are no longer those of basic feasibility; the problems are now those of "scale-up" and of the associated chemical and metallurgical processes necessary to remove excess fission product ash from the core fuel elements and excess new fissionable material from the breeding blanket with high efficiency, so that the "breeding gain" of the reactor will not become a "cycle deficit" due to processing losses.

EBR-II and the PRDC reactor designs are both based on this same EBR-I foundation. In keeping with the difference in objectives of the PRDC organization and of Argonne National Laboratory, these two reactors are designed to achieve very different goals. The PRDC reactor is designed [fol. 4051] to demonstrate the production of as nearly economically competitive power as is possible, and to do so with as few deviations as possible from EBR-I proven principles and technologies. EBR-II, on the other hand, is designed as an engineering test facility; it is a tool for the scientist and engineer, not primarily a power production installation.

The EBR-II facility has been designed to achieve the following objectives:

1. It provides the smallest possible unit with which to test out reactor components scaled up to full production plant size—full scale capacity would be achieved

by multiplying the number of components used, not by increase in the size of the units.

2. It incorporates a chemical and metallurgical pilot plant to test out the chemical separation and metallurgical refabrication processes required to achieve a complete, successful breeding cycle.

3. It includes new departures in sodium cooled, fast reactor technology and will therefore provide operational test of new features designed to simplify construction, improve neutron economy and increase reliability of plant operation.

4. It will yield the information needed not only to provide cost data on its own operation but also that [fol. 4052] required to evaluate costs for other proposed fast breeder reactors and to establish the economic future of reactors of this type, and

5. It will eventually furnish an excellent facility for operational testing of new fuel element designs at high temperature and high neutron and gamma fluxes to determine their thermal and mechanical behavior and their irradiation characteristics.

Although the PRDC reactor has been described in detail in the PRDC License Application, and the details of the EBR-II conceptual design have been published in Geneva Paper #501, a brief comparison of certain major items of the two designs is instructive.

The EBR-II and the PRDC reactor are both sodium-cooled, unmoderated reactors. The EBR-II is an experimental nuclear power plant which will produce 60 megawatts of heat and 20 megawatts of electrical power. Its primary purpose is to develop information relating to nuclear power plants of this type. Since it is a maximum developmental experimental facility, it incorporates flexibility in its design. The PRDC reactor is a prototype central station nuclear power plant which will produce 300 megawatts of heat and 100 megawatts of electricity. Its primary purpose is to demonstrate the feasibility of the production of electrical power to be supplied at the lowest possible

cost and highest dependability to an electrical distribution system.

Although the plants differ in power rating and in primary [fol. 4053] purpose, the PRDC reactor is similar to EBR-II in many significant areas of design and operation. The operation of EBR-II, therefore, will contribute significantly to the proving out of technology applicable to the PRDC reactor.

The PRDC reactor is similar to EBR-II in the following areas:

1. Neutron Spectrum and Flux.

These reactors both operate with a fast neutron flux in excess of 10^{15} neutrons per square centimeter per second. This exceeds present experience by a factor of more than ten, and information obtained from EBR-II will contribute significantly to the PRDC reactor.

2. Power Density (power generation per unit volume of the core).

The PRDC reactor has a core volume of approximately 335 liters in which approximately 270 megawatts of heat is generated, with a resultant power density of approximately 800 kilowatts per liter of volume. The EBR-II core has a volume of approximately 50 liters and produces approximately 50 megawatts of heat with a resultant power density of approximately 1,000 kilowatts per liter of volume. These are very high thermal ratings, and EBR-II experience will be applicable to the PRDC reactor.

3. Fuel Element and Fuel Subassembly.

The PRDC fuel element is very similar to the EBR-II fuel element. They are both small diameter pins of essentially the same size. The PRDC fuel element is jacketed in a thin tube of zirconium, while the EBR-II fuel element is jacketed in a thin tube of stainless steel. These differences are due to different fabrication processes to be employed, but the fuel elements should exhibit similar [fol. 4054] operational characteristics.

In each reactor a cluster of fuel elements is contained in a box or subassembly. In the PRDC reactor 144 elements are contained in a 2.65 in. square subassembly. In the EBR-II, 91 fuel elements are contained in a hexagonal subassembly 2.29 in. across flats. The subassemblies are quite similar in size and should exhibit similar operational characteristics.

4. Primary Sodium Coolant Operating Conditions.

The EBR-II primary sodium enters the reactor at 750F, flows at a maximum velocity of 27 fps through the reactor, and leaves at a temperature of 900F. The PRDC primary sodium enters the reactor at 550F, flows through the reactor at a maximum velocity of 30 fps., and leaves at a temperature of 800F. Although the EBR-II flow conditions are more severe in some respects, the coolant operating conditions are quite similar.

5. Coolant System Flow Arrangement.

Both reactors are cooled by sodium flowing through the core and blanket in parallel. The coolant is supplied to lower plenum chambers which supply coolant to the two regions in the reactor. The coolant flows upward through the reactor to heat exchangers. Upward flow was selected for each reactor to conform with natural connection which is employed to remove fission product decay heat in the reactors after shutdown.

6. Heat Transfer Circuits.

Both reactors employ a non-radioactive secondary sodium system to transfer the heat from the primary sodium [fol. 4055] system to the steam generator, and to isolate the reactor system from the steam system. The plants, therefore, will include similar plant control for over-all operation of these heat transfer systems, i.e., primary sodium, secondary sodium, and steam systems.

The major differences between the two plants pertain to size and purpose. The PRDC reactor is considerably larger, and therefore includes an inherent advantage with

respect to one aspect of operational stability. Because of the larger reactor volume, it will use a lower enrichment fuel, which increases the possibility of a negative Doppler coefficient. Since the magnitude of the Doppler coefficient for EBR-I has been shown to be a very small positive quantity, and since the contribution of U-238 is negative, the over-all coefficient in a more dilute system (lower enrichment) will be still smaller than EBR-I and might actually be negative.

Although the PRDC plant is approximately five times as large as the EBR-II plant, the PRDC system employs three separate heat transfer circuits, while the EBR-II employs only one circuit. Each PRDC circuit, therefore, is less than twice the capacity of the EBR-II circuit, and the engineering development of sodium system components for EBR-II, such as pumps, valves, and flow instrumentation will contribute significantly to the PRDC plant.

The PRDC reactor is designed for a demonstration central station power plant and is based on known technology to the fullest extent possible. The EBR-II is designed for maximum developmental flexibility and includes certain [fol. 4056] features which are perhaps more advanced, and represent a larger deviation from known technology.

Some of the significant differences in plant designs include the following:

1. Reactor and Primary Coolant System Arrangements.

The EBR-II employs a tank-type arrangement for the reactor and primary coolant system in which the reactor and primary coolant system are contained in a single, large vessel and operate submerged in sodium. This arrangement offers certain advantages in compactness of design and in minimizing of some of the problems of sodium plumbing in the coolant system. It requires, however, that sodium system components, such as pumps, heat exchangers, and flow instrumentation operate submerged in high temperature sodium. System components of this type had to be developed to meet these requirements.

The PRDC reactor and primary coolant system are arranged in a more conventional manner with the reactor

contained in a reactor vessel with external piping, pumps, and heat exchangers. This arrangement is similar to all sodium-cooled reactor systems employed to date, and therefore applies the technology already developed.

2. Reactor Control.

The EBR-II reactor employs movable fuel as the method for controlling the reactor. This system permits high neutron efficiency in the reactor, since the control material contributes to the neutron economy of the reactor. Since [fol. 4057] large quantities of energy are being generated in the control units, it is necessary to provide reliable cooling to these units during operation. Providing adequate cooling to these movable units introduces severe engineering problems.

The PRDC reactor employs a neutron poison to effect reactor control. Because neutrons are absorbed parasitically in the control units, this control system reduces the neutron efficiency of the reactor. The heat generation in the control units is quite small and the cooling problem is avoided.

3. Fuel Subassembly Hold-down System.

Both reactors employ high velocity coolant flow upward through the reactor which tends to raise the subassemblies out of the reactor. In the EBR-II, a hydraulic system is employed to provide a resultant downward hydraulic force on each subassembly. This is a rather unique approach to the problem and has not been demonstrated in any existing reactor. The PRDC reactor employs a mechanical system to support and position the subassemblies. Similar arrangements have been successfully employed in other reactors.

4. Sodium Pumps.

The EBR-II reactor system will employ two or three different types of sodium pumps. A variety of pump types is employed (a) to meet the peculiar requirements of sub-

merged operation in the primary system, and (b) to permit operational evaluation of different types of sodium pumps. This latter purpose is in accordance with the over-all objectives of the EBR-II to advance the technology of this type of reactor.

[fol. 4058] The PRDC plant employs only one type of sodium pump which will permit lower costs, because several identical units will be produced and, since only one type of pump will be employed in the entire plant, maintenance will be simplified.

It is clear that much of the information to be obtained from EBR-II and the engineering tests associated with its design and construction are directly applicable to the PRDC reactor. The architect-engineer is already at work on the detailed engineering design of EBR-II. Construction is scheduled to start in May, 1957, and to be completed in January, 1959. Argonne National Laboratory will install the reactor core and associated equipment during the first half of 1959, and reactor start-up is scheduled for June and July, 1959.

In my opinion the extensive research and development work being done in connection with EBR-II, and the experience which will be obtained in its start-up and operation prior to scheduled loading of the core in the PRDC reactor, will supply substantial additional information with respect to the safety of operation of the proposed PRDC reactor. I believe that this information will confirm that the PRDC reactor, or any other well-designed reactor of this general type, can be operated safely for the generation of electric power.

General Reactor Engineering Work Relevant to PRDC Reactor

Two general reactor engineering programs at Argonne National Laboratory have particular relevance to the PRDC reactor project. These are the fuel element irradi- [fol. 4059] ation program and the engineering test work on sodium technology and sodium system components.

The Laboratory has been engaged in studying the thermal, corrosion, and irradiation behavior of fuel materials

and prototype fuel elements since the war days. These studies have included investigations of fabrication techniques and of alloying materials on the corrosion characteristics and on the thermal and irradiation stability of uranium and recently of plutonium and thorium fuel elements. The observed behaviors have been correlated with the physical metallurgical characteristics of the test specimens, and while the field is still far from being thoroughly understood, tremendous strides have been and are being taken in improving this situation. All of this work is directly relevant to the PRDC reactor design and in my opinion has been and is being used to provide that reactor with a satisfactory fuel material.

In addition to the basic study of fuel materials, an intensive program of testing prototype fuel pins in the Laboratory's CP-5 research reactor has been under way for the EBR-II for two years, and tests are just getting under way for the PRDC reactor. In this connection the power level of the CP-5 reactor has already been doubled from 1,000 kilowatts to 2,000 kilowatts in order to double its available flux and a further power doubling to 4,000 kilowatts is under consideration. The higher flux level yields significant information in proportionally shorter time.

Prototype fuel elements for the EBR-II reactor have been under test in the Materials Testing Reactor (MTR) at the National Reactor Testing Station for approximately [fol. 4060] six months.

In my opinion any information needed with respect to the behavior of the proposed fuel elements for both reactors will be in hand well before fabrication of the respective cores is required for either EBR-II or the PRDC reactor.

In the field of sodium technology, the Laboratory has had a continuing program of investigation since 1945. This work was started during the development of EBR-I and has continued essentially without interruption. It has involved the development of both AC and DC electro-magnetic pumps and DC homo-polar generator current sources, of-

mechanical pumps and bearings, of special valves, of pipe heating systems to maintain sodium above its melting point, of flow meters, pressure gauges, level indicators and other necessary instrumentation, and of the many specialized pieces of equipment needed in reactor control, loading, unloading, and the like. It has also involved the development of methods of removing oxygen from the liquid metal system almost completely, and of maintaining purified inert atmospheres above all free liquid metal surfaces. A very large volume of such technological information has been developed not only at Argonne National Laboratory but also at the Knolls Atomic Power Laboratory and at the Atomics International Laboratory. This body of technology has been fully utilized in both the EBR-II and the PRDC reactor designs.

In addition, an approximate half scale mechanical mock-up of the EBR-II has been under continual test using design operating conditions for 16 months in order [fol. 4061] to test all of the necessary special mechanical parts, not only for operability and reliability but also for life span. So far all equipment has functioned perfectly. PRDC has a similar mock-up of their reactor under way for similar tests, as Mr. Amorosi has explained in his testimony.

The Laboratory has also commenced operation of performance tests on full-scale components for the EBR-II. At the present time two full-scale loops are in operation testing an AC electro-magnetic pump in one and a mechanical pump in the other, together with the associated valves and instrumentation. A third full-scale loop to test a DC electro-magnetic pump will be put into operation shortly. These pumps, valves, and associated equipment are of the same general size as those required for the PRDC reactor and the operating information is being made currently available to PRDC.

It is my opinion that most of the information of this character needed to be included in the PRDC Application for License is presently available, and that any remainder will certainly be available prior to loading of the core.

Summary

To summarize, it is my opinion that in the national interest it is important that a full-scale industrial fast breeder be built and tested at the earliest possible time. A research and development laboratory is not generally possessed of the operations philosophy requisite in the design of a plant to be operated as a successful industrial unit, even for demonstration purposes. Such a central station power plant must be designed under the close and [fol. 4062] continuing scrutiny of a competent production organization. In this way the operating and maintenance features essential to low-cost operation can be integrated into the design.

The program and the time schedule of the cognate work in progress at Argonne National Laboratory is such that substantial nuclear information and specialized technological developments required to complete the PRDC project will be provided by the Laboratory before PRDC will be ready to load nuclear fuel into its reactor. While the Laboratory's schedules for the ZPR-III and EBR-II projects are admittedly tight, I can offer assurance that to the extent that they are within the control of the Laboratory we will use our best efforts to meet them.

Finally, I am of the opinion that, on the basis of present knowledge and theoretical physics, a fast breeder reactor of the general type proposed by PRDC can be constructed in such a way that it will be safe for operation in a populated area. I am further of the opinion that, prior to the time the PRDC reactor is to be started up, additional information and operating experience with other reactors of the general type will provide adequate confirmation of this opinion.

[fol. 4064] In evidence January 8, 1957. Tr. 44

BEFORE THE ATOMIC ENERGY COMMISSION

Docket No. F-16

In The Matter Of

POWER REACTOR DEVELOPMENT COMPANY

TESTIMONY OF W. KENNETH DAVIS,
DIRECTOR, DIVISION OF REACTOR DEVELOPMENT,
UNITED STATES ATOMIC ENERGY COMMISSION

Personal Qualifications

My name is W. Kenneth Davis. I am Director of the Division of Reactor Development of the United States Atomic Energy Commission.

I was born in Seattle, Washington, on July 26, 1918. After graduation from high school in Oakland, California, I studied in the College of Chemistry of the University of California for two and one-half years. I then transferred to the Massachusetts Institute of Technology, where I received a Bachelor of Science degree in Chemical Engineering in 1940. Following this, I spent a year attending the M. I. T. School of Chemical Engineering Practice, and another year as Assistant Director of the Buffalo Station of this School. This Station was located in the Lackawanna Plant of the Bethlehem Steel Company. I passed the doctoral examination and was engaged in thesis work but did not complete the thesis after outbreak of World War II, in December 1941. I had given serious consideration to [fol. 4065] work on the separation of uranium isotopes by ion exchange but gave this up in favor of work on gaseous combustion which could be carried out at Buffalo. I received a Master of Science degree in Chemical Engineering in 1942.

In June 1942 I began work for the Research and Development Department of the Standard Oil Company of Cali-

fornia which later became the California Research Corporation, a wholly-owned subsidiary of the Standard Oil Company of California. Here my work was primarily in process design and engineering development although several months were spent in technical service work on an aviation gasoline plant. Principal development projects were on butadiene manufacture and catalytic cracking processes. In May 1945 I was made Acting Group Supervisor, and in May 1946, Senior Research Engineer.

In October 1947 I went to work as a Senior Engineer for Ford, Bacon and Davis, Inc., of New York City. They assigned me to their project in Chicago where they were architect-engineers for the new Argonne National Laboratory facilities. It was my responsibility to work with the laboratory personnel and with the company's engineers to arrive at facilities which would meet the laboratory's requirements. During the course of this work I had a great deal to do with the design of the Chemistry and Chemical Engineering Buildings, and with all the Temporary Laboratory Buildings. I was also responsible for the mechanical [fol. 4066] engineering on some of the Temporary Laboratory Buildings. Among other things, I personally supervised preparation of the first cost estimate for the Experimental Breeder Reactor (No. 1), after reviewing its design. Aside from this type of work, I did not have any direct connection with reactor development work during this period.

In May 1949, I joined the faculty of the University of California at Los Angeles as associate professor of engineering. As chairman of the chemical engineering committee I was responsible for building up staff, initiating curricula and courses, and acquiring laboratory facilities for process and chemical engineering. I was active on a group seeking (unsuccessfully) to obtain a research reactor in the Los Angeles area, or, specifically at the University of California. I also organized the material for a senior-level course in nuclear engineering. During the period at the University of California I engaged in a substantial amount of industrial consulting work. I was promoted to the rank of professor on July 1, 1953, while on leave-of-absence.

In December 1950 I was given leave to join the California Research and Development Company at San Francisco, Berkeley, and Livermore, California. This company was formed as a subsidiary of the California Research Corporation to undertake research and development for the U. S. Atomic Energy Commission in cooperation with the [fol. 4067] University of California Radiation Laboratory. The work done by the company was on the development, design, construction, and operation of very high current linear accelerators. I had the task of hiring, organizing, and directing a group to do research and development work on the target for the accelerator.

In the summer of 1952 the A. E. C. Division of Reactor Development asked the Company to undertake research and development on fast reactors in collaboration with the Argonne National Laboratory. An intensive program was begun on fast reactors under my direction. In October the research and development department was re-organized and I was designated Manager of the Research Division, with the additional responsibility for the fast reactor project as such. Unfortunately, budgetary limitations resulted in the curtailment of this program late in 1953. The Company finally ceased operation in mid 1954.

I joined the Division of Reactor Development of the Atomic Energy Commission in April 1954 as Assistant Director-Technical. In August 1954 I was appointed Deputy Director, I was appointed Director of the Division of Reactor Development in February 1955. In my present capacity I am in charge of and have responsibility for all reactor development work for the Commission, including the Aircraft Reactor Program, the Army Reactors Program, the Civilian Power Reactor Program, and the Naval Reactors Program. Supplementary to these primary activities [fol. 4068] are many research and development activities concerned with the development of reactors, including a broad program for basic development of power reactors, including chemical processing. Among the reactor activities for which I am responsible are those at National Reactor Testing Station, Argonne National Laboratory Los Alamos Scientific Laboratory, Knolls Atomic Power Laboratory, Brookhaven National Laboratory, Oak Ridge National

Laboratory, and others. I specifically have responsibility for the execution of the Commission's fast reactor program.

Because of this administrative responsibility, as well as my previous personal work on fast reactor research, I have followed fast reactor development relatively closely since joining the AEC. I am familiar with this development to date, including specifically the history of the EBR-I, the development and plans for EBR-II, the development and plans for the PRDC reactor, and the work done by the United Kingdom. Although I have not, of course, had an opportunity to read all of the pertinent technical reports, nor to go through the various technical arguments in mathematical detail, I have maintained a continuous technical interest in these reactors to the extent that I am knowledgeable regarding them and their various problems and difficulties, as well as the formal development programs for them.

[fol. 4069] *The History and Development of Fast Reactors*

The United States Atomic Energy Commission's interest in fast reactors goes back to at least 1945, in which year both the Argonne National Laboratory (then called the Argonne Forest Laboratory) and Los Alamos Scientific Laboratory submitted proposals to the Manhattan Engineering District to design and construct one fast reactor by each laboratory. The resulting Los Alamos fast reactor was designated "Clementine", the Argonne reactor was called the Experimental Breeder Reactor, now known as the Experimental Breeder Reactor No. I (EBR-I). I shall discuss each of these in more detail later in my testimony.

In addition to these two reactors, each of which was operated for several years, a number of unmoderated reactors and critical assemblies in which the fissions were due to fast neutrons have been operated in both the United States and England. From experiments with these assemblies, considerable experience and nuclear physics data have been obtained which are of definite assistance in the design of fast power reactors.

A brief description of the fast critical assemblies and reactors already constructed and operated will give an indication of the type of data on fast reactors already available, and will indicate the variety of loadings and

materials used. These machines may be conveniently [fol. 4070] divided into three classes: (a) small highly-enriched critical experiments, (b) flexible critical experiments and (c) fast neutron reactors.

(a) *Small highly-enriched critical experiments.*

Since a large variety of basic nuclear information is most easily and economically obtained from simple uncomplicated experiments a number of such experiments have been run on unmoderated systems.

Lady Godiva. At Los Alamos Scientific Laboratory a bare, spherical critical assembly of roughly 6¾ inch diameter using U^{235} metal enriched to about 90 percent has been in operation since August 1954. This assembly is known as "Lady Godiva." It has no cooling system; thus, its maximum power is limited to 1 kw level during continuous operation.

The assembly consists of three parts which are mechanically assembled by remote control. After the assembly has been put into its spherical shape, delayed neutron criticality is achieved by insertion of two 7/16 inch diameter control rods made of U^{235}.

During remote operation, progress of the assembly operation is shown by closed-circuit television of the set-up, signal lights and selsyn-driven indicators coupled to moving parts of the assembly apparatus. Interlocks are used to assure that the assembly operation is carried out in only a certain safe sequence. Detectors for fast neutrons have [fol. 4071] been used for indicating the assembly power level and also for initiating scram signals for automatic immediate shutdown when the power level is higher than a preset value or if the period is shorter than a preset value. These neutron detectors in conjunction with automatic control circuits, permit power level to be effectively maintained at some set value. Two types of experiments have been run with Godiva. The first type was run with the assembly critical on delayed neutrons. In the second type the assembly was subjected to sufficient reactivity to be critical on prompt neutrons alone; these experiments are called "prompt burst" experiments.

With the delayed critical experiments, much physics data has been obtained. Data on the nuclear properties of materials applicable to fast neutrons have been difficult to obtain. In most cases the techniques which have been most successful in thermal reactors cannot be used for obtaining fast neutron data. Values for nuclear cross sections must generally be arrived at by the more indirect methods of comparing theoretical prediction and experimental results from integral experiments in which several parameters vary.

The effect of several materials on reactivity when placed in different radial positions within Godiva was measured experimentally. These data were used to estimate the transport cross sections of those materials. The transport cross [fol.4072] sections are most important for calculations on fast reactors. The neutron spectrum was also estimated by measuring the results of irradiating various fissionable isotopes and other materials of known energy-dependent cross sections.

In addition to the delayed critical experiments, a number of experiments have been run with Godiva prompt critical. In order to make these prompt-burst experiments possible, a third uranium control rod worth about one dollar in reactivity was added which could be rapidly shot into the assembly. With these experiments it was possible to observe from the neutron level, that, for instantaneous reactivity insertions up to about ten cents above prompt critical, the assembly expanded due to heating sufficiently to reduce the reactivity to below prompt critical in less than a millisecond. In other words, there is a sudden increase of power level to high values but of such short duration that the total energy release is relatively small. During the burst, power build-up of simple assemblies such as Godiva is determined by the prompt neutron lifetime, but the self-limiting characteristics resulting from thermal expansion cause the power build-up rate after the burst to be determined by the delayed neutron lifetimes and the remaining excess reactivity.

[fol. 4073] The experimentally demonstrated self-limiting feature of nuclear assemblies such as Godiva when made just greater than prompt critical are important to increas-

ing our understanding of critical assembly behavior. But also important is the success which has been demonstrated in predicting this kinetic behavior by calculation. Success in correlating calculated behavior with experimental behavior provides greater confidence in calculations which cannot be experimentally observed.

Prompt neutron bursts were used as a source of fast neutrons for irradiation of materials fissionable by fast neutrons such that the yields and decay constants of the delayed neutrons produced by fast neutron fissions could be measured.

Jezebel. A later fast critical assembly at Los Alamos Scientific Laboratory is called "Jezebel." It is a bare sphere made of plutonium operated at low power. The size, construction, and operation of this device are similar to that described for Godiva. Experimental data from it will be used to obtain more accurate parameters for plutonium for use in calculations.

Topsy. "Topsy" was the name given to a small reflected critical assembly which was operated for 8 years at Los Alamos before being disassembled. This uncooled machine consisted of four main subassemblies. These were a 16- [fol. 4074] foot track section, carriage, safety test section, and critical assembly section. The core was assembled on the carriage which was movable along the track section. The core section could then be raised into either the safety test section or the critical assembly section by remote control actuation of the carriage hydraulic lift.

When it was desirable to estimate a safe mass limit for hand-stacking the active material in the core unit of the carriage lift, the core could be remotely lifted into a toroidal water tank safety test unit.

To form a critical assembly, the core was lifted into the hollow reflector which had been stacked on the stationary critical assembly unit. Because of the flexibility available with handstacking, experiments were made with different core densities and compositions. Critical mass was determined for spherical uranium cores of 50 to 90 per cent enrichment and from 50 to 100 per cent density, using a natural uranium reflector. The effect of the use of nickel as a reflector as compared with natural uranium metal

was determined for cores of both 90 per cent enriched uranium metal and uranium hydride. Some experiments were also run using a spherical plutonium core and natural uranium reflector. These experimental data were obtained for the simple geometrical shape such that the results could be [fol. 4075] checked more readily by calculation. The variety of assemblies presented a range of tests for determining the accuracy of nuclear data and reactor physics calculations of reflected, fast reactors.

The assemblies were operated at delayed critical at powers up to 1 kw for the purpose of obtaining data which could be used to derive values of the apparent absorption and transport cross sections of several materials. The method was similar to that used with Godiva.

As in Godiva some data was obtained on the relative fission rates of U-235 and U-238 at various radii. These data were used as indications of the neutron spectra which could serve as checks on the reactor physics calculations for the assemblies.

Zephyr. Still another uncooled, small fast critical assembly, called "Zephyr" and consisting of a plutonium core surrounded by a natural uranium envelope has been in operation at the Atomic Energy Research Establishment at Harwell, England since February 1954.

(b) *Flexible Critical Experiments.*

ZPR-III. When the feasibility of large fast power reactors had been established and designs for them were under way, it became desirable to have available a fast critical assembly for the larger, more dilute fast reactors. Prior to 1955, all critical assemblies had used rather small, highly enriched cores, except for the few exploratory investiga- [fol. 4076] tions of assemblies of lower enrichment and densities using "Topsy" which I have already described.

An uncooled fast critical facility, designed to permit investigations of a wide range of sodium-cooled fast power reactor designs, has been constructed at the National Reactor Testing Station by Argonne National Laboratory. It is known as "Zero Power Reactor III" (ZPR-III). The first loading of ZPR-III went critical in October 1955. I

shall discuss later in some detail the experimental and research program which is planned in connection with ZPR-III, and its importance in providing data relevant to both the design features and safety of start-up and operation of other proposed large power reactors, including the PRDC reactor.

Zeus. Another large uncooled reflected critical assembly facility has been built in England and went into operation in December 1955 at the Atomic Energy Research Establishment at Harwell, England. This facility is known as "Zeus." This reactor is used for studying generally the nuclear characteristics of a variety of dilute large fast reactors using U-235.

(c) *Fast Neutron Reactors.*

Clementine. The first cooled fast reactor was built at Los Alamos Scientific Laboratory in accordance with a de-[fol: 4077] sign conceived in early 1945. Clementine was designed and built primarily to provide a high intensity source of fast neutrons. In addition it permitted study of the control of fast reactors, study of breeding possibilities, and study of power production principles on a limited basis. Valuable information was obtained in these areas.

Clementine was a fast reactor having an average neutron energy in the range of 500 kev with a fast flux of 4.3×10^{12} at 25 Kw. The fuel was plutonium rods, nickel coated, and canned with mild steel, and about 6.5″ long and 0.7″ in diameter. The rods were held in a mild steel cylindrical cage and had a central or peripheral loading. Positions not filled with plutonium were filled with uranium rods to complete the total of 55 positions.

The fuel cage was contained in the reactor pot and was a gas-tight unit. The fuel was cooled by mercury because Na and NaK technology was not developed at that time. The mercury entered through two holes on opposite sides of the bottom of the pot and left through one hole above the center of the cage.

The fuel cage was in the center of a cube of uranium-238 which was the reflector. Four sides of the cube each had a window which was the start of a special test hole or

column. The windows were of thorium, uranium, steel or bismuth. There were other test holes which were tapped to various depths in the reflector.

[fol. 4078] The reflector was cooled by water running through tubes in its outside surface. The reflector cube was surrounded by steel bricks and then lead bricks. This cube was sealed in an air-tight aluminum envelope and flushed with helium.

The remainder of the shielding was composed of layers of steel, masonite, alternating layers of boron plastic and steel and on the outside with an aggregate of concrete and lead shot.

Control was maintained by four vertical rods, two shim and two control, at the corners of the sides of the uranium reflector adjacent to the pot. Their downward movement replaced uranium with boron. Part of the uranium reflector in the shape of a cone below the pot acted as a safety block and was held in by a magnetic clutch.

The mercury coolant was pressurized with helium and circulated by an electromagnetic pump. It was cooled by two water heat-exchangers.

Rupture of a fuel element and subsequent contamination of the mercury with plutonium caused the reactor to be shut down and later dismantled in December 1952, after approximately six years of operation. It had accomplished the research and experimental objectives for which it had been built.

[fol. 4079] *EBR-I.* The second cooled fast reactor and the first nuclear reactor to produce electric power was the Experimental Breeder Reactor No. I (EBR-I), designed and built by the Argonne National Laboratory at the National Reactor Testing Station in Idaho. This reactor was based upon a design conceived by Doctors Enrico Fermi and Walter Zinn in 1945, and was approved as a research and development program by the Manhattan Engineering District late in that year. The Commission approved the EBR-I as a construction project in November 1947. The reactor was located at the National Reactor Testing Station (NRTS) in Idaho in order to permit maximum flexibility in experimentation. The EBR-I went critical in August 1951; electricity was first generated in December 1951.

The principal objectives sought to be accomplished by the construction and operation of EBR-I were testing of the possibility of breeding and the obtaining of data on reactor operation in a high-energy neutron range, and in handling and containing NaK at high temperature.

The core and inner blanket are contained in a stainless steel tank. Two cores have been used and a third is being designed.

The first two cores were composed of stainless steel tubes filled with rods of uranium which had a NaK bond between the uranium and the tubing. A hexagonal stainless steel partition separates the core from the inner [fol. 4080] blanket. The blanket rods were natural uranium and the core rods had 7½″ of highly enriched uranium in the center with 8″ of natural uranium on one end and 4¾″ of natural uranium on the other.

Surrounding the NaK cooled inner blanket and core is the external uranium blanket. The uranium is in the form of stainless steel clad bricks and is air-cooled. This blanket has holes for eight external safety rods and four external control rods. The external blanket is assembled on an elevator that can be raised and lowered to aid in reactor control. At the base of the reactor tank is a safety plug which is part of the external blanket. This plug is held in position by air pressure and in the event of power failure or "scram" moves away from the reactor tank.

Outside the external blanket is a graphite reflector 19″ thick and this is surrounded by about 9′ of concrete. Five beam holes pierce the shielding and reflector for utilization of the high energy neutron flux. Experimental facilities also include a thermal column and "rabbit" hole.

The core and internal blanket of the reactor are cooled with NaK which is circulated through a heat exchanger which transfers heat to a second NaK loop which, in turn, transfers the heat to boiling water in a second heat ex-[fol. 4081] changer. All NaK systems are kept under an inert atmosphere to preclude fire from NaK contact with water or moist air.

As mentioned previously, this reactor produced electricity from nuclear energy for the first time in the world on December 29, 1951. It operated successfully until No-

vember 29, 1955. On the latter date, in the course of a difficult experiment, the EBR-I underwent a power excursion which resulted in damaging the reactor's core; Dr. Norman Hilberry's testimony includes a description of this incident. Action is being taken to replace the damaged core with a new one. Extensive additional research and experimentation is planned with this new core.

(d) *Miscellaneous.*

Three other projects in or related to the fast reactor field should also be mentioned. In 1946 the Manhattan Engineering District contracted with the General Electric Company to contribute to the reactor development program. This contract resulted in the construction of the Knolls Atomic Power Laboratory. This laboratory chose to concentrate its efforts in the exploration of the possibilities of breeding and power production in a reactor whose average neutron energies were in the intermediate range. In 1949 serious consideration was given by the Commission to the construction of a 10,000 kilowatt reactor at [fol. 4082] West Milton, near Schenectady, New York, for the purpose of demonstrating breeding and power generation. However, subsequent information became available which raised questions of technical feasibility of breeding in the intermediate neutron energy range, and this in turn resulted in the Commission's deferring construction on the proposed intermediate neutron breeder reactor. The experimental information and knowledge gained in this program was not lost, however, but was applied to a large extent in the development of the Submarine Intermediate Reactor (SIR). Since 1953 a small number of KAPL scientists have investigated various fast reactor systems such as a fast breeder reactor concept which would utilize plutonium oxide as its fuel. Design studies, supported by irradiation data on plutonium oxide fuels, have indicated that such a reactor system appears feasible.

In 1952 the California Research and Development Company at Livermore, California, with which I was then connected, undertook a fast reactor research and development program, pursuant to a contract with the Commission. In

1953 this work was terminated for budgetary reasons. Prior to such termination, however, many valuable studies were made by the California Research and Development Company on fast reactors. The basic design for the Los Alamos Molten Plutonium Reactor Experiment (LAMPRE) one of the fast reactors now to be constructed for the Com- [fol. 4083] mission was proposed by CR&D in March 1953, based on the earlier work on plutonium alloys done at Los Alamos.

Finally, in addition to the Commission programs, mention should be made of the research and development work which has been under way for several years by Atomic Power Development Associates, Inc. (APDA). I understand that other witnesses will cover this in more detail.

THE AEC FAST REACTOR PROGRAM

Objectives. The development of power reactors where the bulk of the fissions are produced by high energy neutrons—the fast reactor program—is one of the important civilian power reactor development programs of the United States Atomic Energy Commission. The Commission's interest in fast neutron reactors stems from information obtained from experiments and studies which indicate that the fast neutron reactor may have important potential advantages over thermal reactors for the economic development of nuclear electric energy. In my opinion, the fast breeder reactor is one of the promising types for the development of electric power on a commercially feasible basis, and I consider it of considerable importance that the United States demonstrate its continuing leadership in the peaceful application of nuclear energy by successfully developing this type of reactor. The proposed re- [fol. 4084] actor of Power Reactor Development Company (PRDC) can, I believe, be an important part of this demonstration, and it is on this basis that the Commission has accepted PRDC's proposal as a basis for negotiation of a contract with PRDC in connection with the Commission's Power Reactor Development Program. I shall discuss this further at a later point in my testimony.

Some of the advantages of the fast neutron reactor over the thermal reactor for the generation of electric power are:

1. Because the parasitic or non-productive capture cross section of uranium-235 and plutonium is low in the fast neutron spectrum, neutrons are utilized more efficiently than in thermal neutron reactors. This contributes to the possibility of successful breeding.

2. In fast neutron reactors, the chain reaction is sustained by neutrons whose energies range from several million electron volts down to about fifty thousand electron volts. In this energy region significant quantities of U-238 will fission and this amounts to a bonus of over 10 per cent in total fissions.

3. The decrease in reactivity due to the build-up of fission products is nearly negligible in a fast reactor.

[fol. 4085] 4. Fast reactors require less excess "reactivity in hand," and by proper design can probably be operated with less than $1.00 excess reactivity, whereas thermal reactors usually require over $10.00 excess reactivity in the clean-cold condition in order to reach operating temperatures and to override xenon poisoning. However, fast reactors are generally more sensitive to changes in dimensions and geometry distortions than are thermal reactors.

5. Because the neutron capture cross sections of structural materials is low in the fast neutron spectrum, there is considerably more latitude in the selection of reactor structural materials.

6. Fast reactors are characterized by small volumes and compactness. As technology improves, it appears reasonable to hope that fast reactor plants will be relatively inexpensive to construct.

Coupled with these established advantages, the fast reactor presents some unique problems which are receiving extensive further study. These problems may be broadly classified as having to do with (i) engineering, (ii) nuclear

operating stability, and (iii) possible results of a credible meltdown accident.

[fol. 4086] (i) The principal novel engineering problems are:

1. Because the development of equipment to handle sodium and sodium-potassium is not as far advanced as equipment for water-reactor systems, progress in the development of fast reactors has been largely limited by the rate at which satisfactory sodium and potassium-sodium components could be developed, since hydrogenous coolants cannot be used in fast reactors.

2. The necessity for removal of very large amounts of heat from the small core volume of a fast reactor necessarily presents design problems. Because the heat released per unit volume is necessarily higher in fast than in other types of reactors, the problem of adequately cooling the core is also necessarily greater.

(ii) Problems having to do with the nuclear operating stability of fast reactors have, I believe, been over-emphasized by the EBR-I meltdown incident, which I shall discuss later. Additional experimental work is nevertheless required in order to show the best way in which they can definitively be solved. These problems may be considered generally as:

[fol. 4087] 1. The factor or factors responsible for the prompt positive temperature coefficient and for the nuclear instability found in EBR-I must be positively determined.

2. The magnitude and sign of the Doppler effect on fast reactors must be quantitatively determined.

3. Pile-oscillation experiments must be performed on fast reactors to demonstrate conclusively that such experiments are feasible and are capable of being interpreted to give reliable information.

(iii) The third group of problems presented by fast reactors is found in the possible results of a meltdown acci-

dent. As both Dr. Hilberry and Dr. Bethe have pointed out in their testimony, the actual effect of the EBR I meltdown in the presence of sodium provides promise that this problem may not be as severe as some may have expected. The fact that fast reactors contain substantially more enriched U-235 than do thermal reactors of equivalent power, the fact that these reactors are somewhat more sensitive to geometry than thermal reactors, the fact that reactivity may be substantially increased by the accidental introduction of a moderator, and the fact that on meltdown of a fast reactor the formation of a critical mass might result in greater energy release, nevertheless make further research and experimentation in this field desirable. The [fol. 4088] problems to be dealt with in such research and experimentation principally are:

1. The probability and consequences of the formation of a critical assembly during a fast reactor core meltdown. To be explored by both theoretical studies and by experiments to determine the manner in which fuel elements melt during a power excursion.

2. The probability of containing the forces released in a maximum credible accident for fast reactor designs, including those for EBR-II and the PRDC reactors.

The first group of these problems—the engineering problems—although related to safety, concern most directly the satisfactory operating characteristics of the reactor. These problems have been and are being studied in both the AEC and PRDC programs. I also am confident that the results of these programs, including the substantial additional experience to be obtained during start-up and operation of EBR-II, will provide important further data in this field. Additional assurance should be derived from the operating experience to be obtained with the sodium-cooled fast breeder reactor being built by the British at Dounreay, Scotland, to which I shall refer later.
[fol. 4089] The second and third groups of problems—nuclear operating stability and possible meltdown—are being explored in various phases of the AEC fast reactor

program. In addition, valuable information and practical operating experience is expected to be obtained from the British. In my opinion, these programs should at least point the way to satisfactory solutions to these problems prior to the scheduled completion date of the proposed PRDC reactor.

I shall now discuss in more or less detail certain phases of the AEC fast reactor program, with emphasis on those activities which have particular bearing on fast reactor safety. This program will be considered in several parts:

(1) Reconstruction and continued operation of EBR-I by Argonne National Laboratory with new cores, to explore the temperature coefficient and stability questions raised by prior experiments.

(2) Design, construction, pile-oscillation testing and operation of a new medium-power fast breeder reactor (EBR-II) by Argonne National Laboratory.

(3) The zero power reactor program and related work, a theoretical and experimental program directed [fol. 4090] towards investigating and improving the characteristics and behavior of fast reactors under normal and transient conditions, including exploratory research and development work on new types of fast reactors and thermal-neutron fast neutron reactors, and basic work on physical constants and properties of importance in all fast reactors.

(4) The design, construction, testing and operation of several new plutonium-fueled fast reactors, the Los Alamos Molten Plutonium Reactor Experiment (LAMPRE) by Los Alamos Scientific Laboratory.

(5) Theoretical and experimental work with respect to the accidental formation of critical assemblies in a fast reactor by meltdown.

(6) Theoretical and experimental work with respect to containment of maximum credible reactor accidents.

(1) *EBR-I.* The design, operation, testing, and meltdown of EBR-I are described by Dr. Norman Hilberry in his

testimony in ~~this proceeding~~. I agree that in a meltdown of a sodium-cooled fast reactor core a good possibility exists that the reactor would terminate an excursion even [fol. 4091] if its safety circuits failed; that a probable cause of the instability and positive temperature coefficient, exhibited by EBR-I, and its eventual meltdown, was inward bowing of the fuel elements; and that experiments with the proposed Mark III core, which is designed for maximum core rigidity which can be varied for experimental purposes, should, in any event, give a definitive answer as to whether this was the cause of the abnormal behavior of the Mark II core.

I would also like to note that, aside from the inward bowing of fuel elements, two other causes have been suspected to account for the instability noted in the Mark II core under certain operating conditions. It was at first thought that the temperature of the sodium entering the reactor varied, and that this variation was to blame. Experimental measurements proved, however, that the sodium inlet temperature was essentially constant. It was also theorized that the Doppler effect, that phenomenon consisting of an increase in reactivity when the temperature of U-235 is increased, was the cause. However, theoretical calculations indicated that the Doppler effect was too small to account for the instabilities, and this conclusion has now been confirmed by the first of a series of experiments conducted by Argonne on a mock-up of the EBR-I core in a zero power reactor (ZPR-III).

[fol. 4092] That bowing of the fuel elements was probably responsible for the observed positive prompt temperature coefficient is corroborated by the difference in construction between the Mark I and Mark II cores. As Dr. Hilberry points out, the physical construction of the former permitted potentially far less bowing than that of the latter. This fact, taken together with the fact that only minor instability was observed in the Mark I core as compared with the Mark II core, indicates the likelihood that bowing and instability are closely linked. It is true that the Mark I core, unlike the Mark II, was operated primarily at or near ~~full~~ power with high coolant flow, but there is, ~~in my~~ opinion, a reasonable probability that comparable nuclear instabilities would

have been observed in the low-power start-up period, which lasted about a month before the reactor with the Mark I core attained full power. Since none were observed, it seems probable that the Mark I core was in fact more stable than the Mark II.

In the proposed new Mark III core, the annular spacing within the sheathing of each fuel pin will be completely eliminated, since the uranium will be coextruded with zirconium to produce a metallurgically bonded sheathing. In this respect the design is similar to that proposed for the PRDC reactor. Furthermore, as Dr. Hilberry indicates, the Mark III core has been designed so that both the type of [fol. 4093] coolant-flow and the physical limitations on potential bowing can be varied. This flexibility will permit exhaustive experimentation and analysis regarding the factor considered at this time most likely to have caused instabilities, namely bowing of fuel elements.

Oscillator experiments will be conducted on the Mark III core by mechanically varying given amounts of excess reactivity up to about 4 cents, in a sinusoidal pattern as a function of different given frequencies, ranging from about 0.01 to 10 radians per second. Because the amount of excess reactivity can be a figure chosen close to zero, I believe these experiments can be conducted with a high degree of safety. After the pattern of behavior of the reactor becomes known, greater amounts of excess reactivity can be used if required. The resulting power oscillations will be compared to the pattern of the reactivity insertions with particular reference to amplification and phase shift of reactor-output power to reactor-input reactivity. The final results of these tests on the Mark III core will be compared with the results previously obtained in oscillator experiments on the Mark II core in which finite reactivity insertions produced divergent power oscillations in what appeared to be a resonance response. Of course, other kinetic tests are also being considered.

[fol. 4094] Following completion of experimentation on the rigidized Mark III core, the fuel element pins will be modified so that free bowing of the fuel elements can occur. Pile-oscillator tests, and possibly other tests, will again be conducted with the Mark III core as thus modified so that

the effect of free fuel element bowing upon nuclear instability can be studied. I expect that the Mark III core under these new conditions may exhibit resonances in pile-oscillator tests, although probably of lesser magnitude than those found within the Mark II core, since bowing within the sheathing will not be possible. This should determine whether bowing was the cause of the EBR-I positive prompt temperature coefficient.

The estimated schedule for the EBR-I with the Mark III core is as follows:

In Operation	August 1957
Preliminary Pile-Oscillator Data Available	October 1957
Tests Finished on Rigid Core	April 1958 or earlier
Modification of Rigid to Non-Rigid Core	May 1958
Pile-Oscillator Tests Finished on Non-Rigid Core	December 1958

When these experiments on the stability and fuel element bowing problems have been completed, the Division of Re- [fol. 4095] actor Development and the Argonne National Laboratory will consider loading the EBR-I with a plutonium or plutonium-uranium core which could be available in early 1959. Operational tests on such a Mark IV core of plutonium will probably require one year. There is a possibility that during 1960 the EBR-I would be made into a facility for training reactor engineers and operators.

The results of this proposed EBR-I program should in my opinion serve to isolate the cause of the instabilities which were observed with the Mark II core and indicate appropriate design and engineering correctives which can definitely eliminate them. Before leaving EBR-I, however, I want to emphasize again that the meltdown occurred during an experiment in which extremely abnormal operating conditions were simulated. The fact that EBR-I had previously been operated over considerable periods of time under normal operating conditions, during which time it

showed no significant positive temperature coefficient or instability, is highly important. Indeed, with the exception of the Submarine Thermal Reactor, no other AEC power reactor has operated for so long a period. This experience lends strong support to my view that fast reactors can be so constructed and operated as to be practicable and safe.

(2) *EBR-II.* The Experimental Breeder Reactor No. II will be a fast reactor of advanced design based upon experience gained from the EBR-I. EBR-II will be a two-region reactor which will use uranium-235 fuel at the be- [fol. 4096] ginning, and plutonium alloys as fuel after about two years of operation. It will be located at the National Reactor Testing Station in Idaho.

Upon completion of construction and installation of equipment in EBR-II in 1959, pile-oscillation tests will be conducted over a period of months to determine the presence or absence of undesirable resonances. These experiments will provide further indication also, as to both the safety of such tests and the reliability of results which are obtained from them. If any indication is received of the presence of unsafe resonances, operation of the reactor at significant power levels will be postponed until the cause of these resonances is isolated and the resonances are eliminated by appropriate design modifications. Since by that time we will already have the results of the EBR-I program with the Mark-III core, and will have incorporated into the final design of EBR-II any changes indicated by the results of that program, I am confident that no resonance problem will be found which cannot feasibly be solved.

I believe there is merit in conducting pile-oscillation tests in determining the characteristics of fast (and also of thermal) reactors in the start-up or pre-operational testing phases. The Commission has had considerable experience in conducting such tests.

[fol. 4097] A comparison of the proposed operating parameters of the EBR-I, EBR-II, and Power Reactor Development Company's fast reactor, is given in Exhibit 1, incorporated in and made a part of this testimony. In addition to the fact that the EBR-II will have somewhat higher performance specifications than the PRDC reactor, it will also use more advanced ideas in design, such as:

A. Spent fuel will be reprocessed by pyrometallurgical techniques.

B. Fuel elements will be fabricated by remote procedures.

C. Reactivity changes and shut-down will be accomplished solely by moving fuel.

D. All radioactive "hot" reactor components will be in one container which also is the sodium containing tank.

E. The largest sodium pumps thus far developed will be used.

Although the EBR-II is not a prototype of any specific full-scale power reactor, the information and experience gained from the EBR-II will be of value to fast reactor designers insofar as many components are concerned; *e.g.*, fuel elements, pumps, valves, seals, instruments, etc. The stability characteristics, general performance, and operating characteristics of the EBR-II will be investigated by [fol. 4098] pile-oscillator experiments and will be of great importance in evaluating other fast reactor designs.

The Division of Reactor Development is exploring those practical methods by which the design and construction of the EBR-II can be accelerated. At the present time the Argonne National Laboratory estimates that the EBR-II will be ready for operation about two and one-half years after the Architect-Engineer began the detailed design which was in November, 1956. Its present estimated schedule is as follows:

1. *Construction Phase*

A. A-E Contractor on Duty	November 1956
B. Construction to Start	May 1957
C. Construction Completed	January 1959
D. Equipment Installed	March 1959
E. Reactor Critical	June 1959

2. *Operation and Testing Phase*

A.	Pile-Oscillator Tests	July-December 1959
B.	Steady-State Performance	All of 1960
C.	Steady-State Performance Using Plutonium-Uranium Fuel	All of 1961 and thru 1962

Integrated into the research and developmental activities of the EBR-II is a concerted effort to improve component [fol. 4099] design, efficiency, and general reliability of mechanisms which must be in contact with molten sodium at high temperatures. The ANL Liquid Metal Facilities with the scaled-down mechanical model of the EBR-II, and the three hot sodium loops using the largest capacity sodium pumps thus far developed, have done much to advance the knowledge of the use of sodium as a reactor coolant.

(3) *Zero Power Reactors:* Unlike many other power producing machines, a reactor can operate at essentially zero power with relatively similar behavior and with many of the operating characteristics which it would have at a much higher operating power. Consequently, many of the important nuclear data needed for the development of full-scale reactors can be obtained by using a zero power reactor which is less expensive to construct, of greater flexibility, and more convenient to operate. However, such facilities usually do not utilize power-reactor fuel elements, and do not operate much above room temperature nor contain significant amounts of fission products. I consider two such reactors, Zero Power Reactor No. 3 (ZPR-III) and Zero Power Reactor No. 5 (ZPR-V) to be important parts of the over-all fast reactor program because the ZPR-III is a flexible instrument which will be used to provide important nuclear information to fast reactor designers, and because the ZPR-V is a coupled fast-neutron–thermal-neutron assembly which will increase our understanding of such coupled systems.

[fol. 4100] ZPR-III is a fast zero power reactor located at the EBR-I site at the National Reactor Testing Station at Idaho. It is so designed that nuclear core configurations for specific fast power reactors can be studied and analyzed so

that the final design of a fast reactor can be done with a minimum of uncertainty. ZPR-III has been accurately described in detail in the unclassified Argonne National Laboratory Report ANL-5512, and is also discussed by Dr. Hilberry in his testimony in this proceeding. I will therefore only present its general role in the EBR-II and PRDC design programs.

A comprehensive series of experiments is already under way in ZPR-III to determine the sign and magnitude of the Doppler effect in a fast neutron spectrum. The first such experiments, completed by Argonne National Laboratory in early 1956, showed that the Doppler effect was too small to be measured with the small sample used in the existing experimental setup. Further experiments were started in ZPR-III on October 29, 1956, using improved experimental procedures. Most of this experimental work has now been completed, and final results will be available as soon as the data have been thoroughly analyzed. Preliminary analysis indicates, however, that the room temperature coefficient of the Doppler effect of U-235 in the EBR-I assembly is $+2\text{x}10^{-6}\ \frac{\text{delta } k}{k}$ per degree centigrade. As Dr. Hilberry [fol. 4101] has stated in his testimony it is clear from this that the Doppler effect could not have been responsible for the positive temperature coefficient in EBR-I, and is actually less than that predicted by earlier calculations.

Another important use to which ZPR-III will be put is in determining the amount of uranium-235 necessary for any specific fast reactor to become critical. Multigroup calculations are not accurate enough to determine the exact critical mass associated with a specific design, but such calculations usually give an estimate of sufficient accuracy to permit a preliminary core and blanket design. However, for the final reactor design, more accurate critical mass data must be obtained so that one can have the assurance that the reactor can be made critical at start-up, and with a core size close to that desired.

By analyzing a simulated reactor core and blanket geometry of a specific reactor design, information can be obtained to minimize excessive neutron leakage in the final core and blanket design, as well as to optimize and deter-

mine the reactivity worth of the control system. Other nuclear parameters of importance can also be determined.

The ZPR-III will be used in experiments to determine the worth of control components which will be incorporated in the designs of both the EBR-II and PRDC pile-oscillation mechanisms.

[fol. 4102] The ZPR-V, located at Lemont, Illinois, can be considered to be a reactor within a reactor: It consists of a five-foot diameter steel tank in which the uranium fuel assemblies and the control rods are placed. In the center of this tank is the fast neutron section and surrounding this section is the slow neutron section. The thermal neutron section contains light-water as the moderator, whereas the fast neutron section cannot have a moderator. The ZPR-V is unique in that the nuclear chain reaction occurs in both regions although they have entirely different nuclear properties.

This two-region critical assembly utilizes the best features of thermal and fast critical assemblies, the ease of control of the thermal section and the improved nuclear performance of the fast section. About ninety-five percent of the neutrons are formed in the fast section and five percent are formed in the thermal section. The reactivity of the entire assembly is varied by controlling the thermal neutrons.

The objectives of the ZPR-V program are to determine the feasibility of scaling up the experiment to a power producing pilot model and of obtaining additional nuclear data in an area in which further exploration is desired.

(4) *LAMPRE.* Although fast breeder reactors of the general type represented by EBR-I, EBR-II, and the proposed [fol. 4103] PRDC reactor are among the more promising for the eventual commercially feasible production of electric power, the fact that they use solid fuels is an economic drawback. Solid fuels limit burn-up of fissionable material because of irradiation damage, and increase the total cost of operations because of the necessity for fuel element fabrication and reprocessing. Looking toward an even more advanced reactor, therefore, consideration has been given to the development of fuels which are non-solid and can operate at very high temperatures; also, it is likely that the heat transfer rates can be increased with molten

fuels. Investigations by Wensch and others, at the Los Alamos Scientific Laboratory in 1949, indicated that a low melting eutectic existed between plutonium and nickel. Other subsequent investigations revealed that lower melting eutectics existed between plutonium and cobalt and plutonium and iron. The compositions of the eutectics are high in plutonium, near 88 atomic percent plutonium, and have melting points ranging from 405°C for plutonium-cobalt, 412°C for plutonium-iron to 465°C for plutonium-nickel. The melting point of plutonium is about 637°C.

In early 1955, the Division of Reactor Development assumed programmatic responsibility for the guidance and direction of the Los Alamos Reactor program, and recommended that serious consideration be given to a fast reactor employing a molten fuel. At a previous time the [fol. 4104] Brookhaven National Laboratory had undertaken a program to develop and evaluate a thermal neutron reactor using liquid metal fuel, a solution of uranium in bismuth.

Preliminary investigations have indicated that one of the most difficult technical requirements to meet in the Los Alamos Molten Plutonium Reactor Experiment, designated LAMPRE, is one of providing suitable containment for the molten plutonium alloy. The containing metal must be amenable to fabrication, be inert to molten plutonium and coolant, and have a high melting point. Plutonium is difficult to contain by most metals as it is metallurgically rather active.

Of the many materials investigated by the LASL, tantalum appears to be a promising metal for use in fabricating the core structure. Small-scale laboratory tests reveal that tantalum undergoes insignificant amounts of corrosion when heated in contact with molten plutonium at temperatures up to 1000°C for several thousand hours. Samples containing plutonium in tantalum have been irradiated in the MTR in order to determine the influence of fission products and irradiation upon tantalum stability. The results obtained have been encouraging. Other activities concern reactor control techniques to be used, pyrometallurgical reprocessing of plutonium, and improving sodium coolant properties. The last mentioned activity is important as the

[fol. 4105] sodium coolant must contain very little oxygen, much less than that used in EBR-I, EBR-II, and SRE (Sodium Reactor Experiment) because tantalum is very reactive with even traces of oxygen at high temperatures.

As presently planned, the molten plutonium fast reactor program will be divided into three phases; the design, construction and short-term operation of an experimental reactor, LAMPRE-I; the design, construction and longer term operation of a larger more advanced reactor experiment LAMPRE-II, and possibly the design, construction, and operation of an electrical power producing reactor, LAMPRE-III.

The significance of this type of fast reactor to the solid fuel fast reactor probably lies most in the information on molten alloys that will be obtained. This information should be of importance in analyzing the results of a melt down of a solid-fuel fast reactor. Information obtained on the properties of materials with respect to molten alloys will also be of interest in designing safe solid-fuel fast reactors. Considerable experience will be obtained in techniques to pyrometallurgically reprocess plutonium and in handling large quantities of irradiated plutonium containing fission products.

(5) *Meltdown Calculations and Experiments.* The formation of a secondary critical assembly, resulting from a melting down of a fast reactor core, is a theoretical possibility. [fol. 4106] Because a fast reactor must use enriched fuel, its core can become more reactive if the fuel is brought together more intimately during the meltdown. Although detailed plans are as yet largely unformulated, this represents an area of technology in which important further research and experimentation work will be undertaken.

The melting down of a fast reactor core could be caused by a stoppage of the coolant flow or by a significant increase in reactivity through accident or by any event in which the prompt and decay heat is not removed at a rate sufficient to prevent the melting of the fuel. It is difficult to see how any credible accident to a properly designed fast reactor could in fact lead to a melting down of the reactor core because one can prevent by physical means the addition of reactivity at too rapid a rate, and a complete coolant stop-

page can be made most unlikely. The meltdown of the EBR-I core resulted from an excursion during the performance of a very difficult experiment in which the coolant flow was purposely stopped and other devices purposely bypassed. There would be no purpose in conducting such an experiment on a commercial-size reactor. Nevertheless, melting down of a fast reactor core under some hypothetical conditions may increase the reactivity, at least in theory, at a rate sufficiently high to produce an explosive force, but a force many orders of magnitude lower than what is usually associated with even a very small atomic explosion.

[fol. 4107] The AEC program contemplates that idealized models can be evaluated analytically using credible rates of reactivity and the other nuclear parameters which may be associated with the assembly mechanisms considered. Among the various assembly mechanisms which will be considered are the following:

1. Uniform melting and free-fall collapse of the core.
2. Non-uniform melting of the core; *i.e.*, starting first at high neutron flux zones and proceeding to lowest flux zones, and free-fall collapse following melting pattern.
3. Similar melting as described in item 2, but the free-fall collapse to follow an arbitrary pattern seeking a maximum rate of reactivity insertion.
4. The mechanism of terminating a critical assembly will also be studied considering different limiting conditions such as boiling point temperatures of sodium and uranium.

I recognize that theoretical analyses and laboratory investigations of core meltdowns may diverge from the actual behavior of a real reactor core meltdown. For example, meltdown calculations may not take into consideration the decrease in density which in fact occurred when the [fol. 4108] EBR-I core melted down in the presence of sodium, the decrease in density being so great that the critical mass requirements of the core may have increased by a factor of four. However, this decrease in density probably occurred after the reactivity was reduced by the movement of the cup. In the near future the ANL expects

to have a thorough report on EBR-I core showing (1) a material balance, (2) cause of void formation, and (3) results of core behavior upon initiating the termination of an excursion.

At the present time the ANL is planning to conduct experiments in which uranium will be melted, using electricity, and subsequently dropped into molten sodium. The purpose of such experiments will be to determine whether the volatilization of the sodium will result in forming sodium vapor bubbles or voids in the uranium with an apparent decrease in density similar to what was observed in the EBR-I Mark-II core. If it is found that the density of uranium is decreased in a reproducible manner, then the possibility exists that fast reactors using sodium coolant could terminate an excursion by such a mechanism.

The Argonne National Laboratory and Los Alamos Scientific Laboratory are working in a joint effort on studying many of the problems involved in evaluating what happens when a fast reactor core melts down. Because this joint program has been initiated only recently and because [fol. 4109] of the difficulties involved in investigating such a complex problem, I cannot present a specific programmatic outline at this time, but I can state that both Laboratories are now engaged in developing such a program.

(6) *Containment Calculations and Experiments.* Methods of calculation which have been developed can be used to estimate the energy yields per unit mass of core materials. These estimates can then ~~be used to predict~~ the structure necessary to contain the nuclear-produced forces.

A program of calculation and experimentation is being undertaken by various organizations pursuant to contracts with the Commission, to explore further the containment problem as outlined below:

(a) *Naval Ordnance Laboratory.* This organization, located at White Oak, Maryland is presently engaged in the investigation of the general problem of containing adequately both thermal and fast reactors. Size factors, temperatures of the systems, and rates of loading from simulated excursions are under study. A

specific program is to evaluate the probability of containing a commercial-size fast reactor during a postulated excursion, with particular reference to evaluating the influence of the considerable amounts of shield- [fol. 4110] ing materials surrounding the core. As an example of a full-scale reactor, the design specifications of the proposed PRDC reactor are to be used by NOL as parameters for portions of these studies, so that substantial containment data with respect to this particular design should be obtained under this program. In this respect shielding materials may dampen or absorb significant amounts of the released forces, but the possibility must also be explored that fragments spalled from the shielding (missiles) could move with high velocities and even penetrate the container. I hope that preliminary results of this specific work will be available before June 1957.

(b) *Aberdeen Proving Ground.* The Ballistics Research Laboratory of this organization is conducting a theoretical and experimental study of outer containment vessels to determine the applicable design criteria which will provide the desired degree of safety. This includes blast and shock phenomena, but not missiles.

(c) *Armour Research Foundation.* This group is investigating shock wave characteristics as related to simulated reactor excursions. Effort is also being expended in a development program to design blast shields for reactors.

[fol. 4111] (d) *Oak Ridge National Laboratory.* Since it is important to know the distribution and characteristics of fission products released in a serious reactor excursion, this laboratory is making experimental melting investigations on sections of irradiated fuel elements.

Conclusion—The Fast Reactor Safety Program.

Considerations of the safety of fast reactors have been an important and integral part of the fast reactor program of the Commission since its very beginning. A substantial

portion of the program which I have just described is directed toward perfecting and proving the design of fast breeder reactors so as to render them safe for operation near populated areas. The following is a summary and schedule of those aspects of the AEC fast reactor program heretofore described which are particularly pertinent to reactor safety:

Objectives	*Estimated Completion Date*
1. Availability of new Doppler Effect Data from ZPR-III in Final Report	March 1957
2. Complete report on Melted-down ERB-I Core	March 1957
3. A. ERB-I Placed in Operation	August 1957
B. Gross Data Obtained from Pile-Oscillator Tests	Mid-October 1957
C. Rigidized Core Experiments Concluded	April 1958
[fol. 4112]	
D. Non-Rigid Core Experiments Completed	January 1959
E. EBR-I Loaded with Plutonium-Uranium Core	February 1959
F. Steady Power Operations Until	June 1960
4. A. EBR-II Critical	June 1959
B. Pile-Oscillator Tests Completed	December 1959
C. Steady Power Testing and Operation	All of 1960
D. Steady Power Testing and Operation Using Plutonium-Uranium Fuel	All of 1961 and 1962
5. LAMPRE-I Begins Operation	About 1958

Objectives	*Estimated Completion Date*
6. A. New Analysis of Melt-Down Problem by ANL with Assistance of LASL	Due Fall 1957
B. Non-Nuclear Melt-down Behavior of Fuel Elements by ANL to Determine if Sodium Volatilization Decreases Uranium Density. Preliminary Data Due	July 1957
C. If necessary, Nuclear Fuel-Element Melt-Down Facility (Borax Type) will be Initiated at NRTS. Borax Unit will supply Neutrons to Melt a Fast Reactor Fuel Element so that Melting Mechanism can be Observed	Undetermined.

The scientific effort being expended in the fast reactor program has increased with each passing year. In fiscal year (FY) 1957 there was a significant increase in scientific [fol. 4113] manpower because the LASL entered into the program. It is expected that the direct scientific manpower will increase up to FY 1960. The indirect scientific manpower, devoted to specialized disciplines related to fast reactor systems such as plutonium fuels technology, pyrometallurgical reprocessing research and development, and others will also increase by a large percentage.

Direct scientific and engineering development manpower is shown for FY's 1956, 57 and 58. The effort shown below for FY 59 and 60 is based upon past experience and reflects what is considered to be an orderly growth rate.

Scientific Manpower

	FY56	*FY57*	*FY58*	*FY59*	*FY60*
ANL	80	94	135	150	162
LASL	36	43	51	58	65
Total	116	137	186	208	227

The United Kingdom Fast Reactor Program

In February 1956, the Commission and the United Kingdom Atomic Energy Authority approved the exchange of detailed information on fast reactor technology under Section II.A.2 of the existing Agreement for Cooperation between the two countries. There have been several interchanges of visits between the U.K.A.E.A. and ourselves in which valuable design data have been exchanged.

[fol. 4114] I visited the U.K.A.E.A. most recently in October 1956 and had the opportunity to inspect their fast reactor program which is being pursued in a vigorous manner. It was also interesting to observe that about one-third of their total research, development and design manpower available to the entire power reactor development program is engaged in the fast reactor program. In order to stress the importance placed by the U.K.A.E.A. in their fast reactor program, I will describe it briefly below:

The first fast experimental reactor which was designed and built as part of the British program is known as the "Zephyr." It is one of the uncooled small fast critical assemblies to which I referred briefly early in my testimony. It is located at the Atomic Energy Research Establishment (A.E.R.E.) at Harwell, and became critical on February 5, 1954. The British have stated that they intend to use future core loadings of different compositions to extend the data so as to obtain a more comprehensive quantitative knowledge of the nuclear processes which take place in plutonium-fueled fast reactors.

The second experimental fast reactor to be built by the British is named "Zeus," and is also located at Harwell. It is a full-scale, uncooled flexible critical mock-up of the 60-megawatt fast reactor which is to be built at Dounreay, Scotland. It was put into operation on December 22, 1955.

[fol. 4115] Zeus was built primarily to study the neutron characteristics, critical conditions and control calibrations for the Dounreay reactor. The British plan, however, to use Zeus for studying more generally the nuclear characteristics of a variety of dilute large fast reactors, and the data to be obtained from these studies will provide a valuable check on the work to be done along similar lines

in this country. Considerable information on Zephyr and Zeus has been published in the Geneva papers, "Proceedings of the International Conference on the Peaceful Uses of Atomic Energy, 1955."

The first U.K. fast reactor which will produce power is located at Dounreay, in Northern Scotland. It is expected to produce 60 thermal megawatts, which will approximate the output of the EBR-II, 62.5 thermal megawatts. It is a two-region reactor in which the axial blanket is integrated with the core, as in the U.S. designs. Sodium is the coolant, and an intermediate sodium coolant loop is employed. A steel sphere is used for containment.

It is estimated that the Dounreay reactor will become critical in midyear 1958, which will be about one year ahead of the EBR-II and some two years before the PRDC reactor goes critical. In the early part of the Dounreay operation, pile-oscillation experiments will be conducted to determine the inherent kinetic stability of the reactor.

[fol. 4116] I am confident that the Dounreay reactor will provide valuable and important information on such items as the Doppler effect, temperature coefficient, nuclear stability, etc., as well as a test of various engineering designs and the results of actual operating experience. In addition, we shall have the benefit of British research and experimentation, including safety studies, being conducted prior to start-up of the Dounreay reactor. All of this will permit our groups to modify their designs or undertake additional research and development work if necessary.

Relationship of AEC and PRDC Programs

In order to stimulate active industrial participation in the development of reactors, the Commission announced its Power Demonstration Reactor Program on January 10, 1955. The purpose of this program was to provide a framework through which industry could undertake, with Commission assistance and participation, the development, design, construction, and operation of demonstration nuclear power plants. This invitation to industry recognized that many industrial organizations interested in nuclear power did not have access to sufficient research and development

facilities and lacked the resources to absorb all of the excess costs of development, design, construction and operation of demonstration nuclear power plants.

[fol. 4117] In response to the Commission's invitation, the Power Reactor Development Company (PRDC) submitted a proposal to the Commission on March 30, 1955, in which the Company proposed to construct and operate a commercial-size fast reactor power plant, and requested that the AEC undertake certain research and development work having a direct bearing upon the safety and feasibility of the proposed PRDC reactor. This proposal has been accepted as the basis for negotiation of a contract between PRDC and the AEC under the Power Demonstration Reactor Program. An outline of such proposed research and development work, as it has been tentatively accepted by both parties as a basis for further negotiation, is contained in Exhibit XX to the PRDC Application for License.

From its contractual relationship with Atomic Power Development Associates, Inc. (APDA), PRDC already has available to it the resources of a research and development program in this field which has been under way for some five years, with particular emphasis on the design and materials phases of the fast reactor program.

If this contract is executed, therefore, PRDC, as well as any other interested groups, will have available to them the results of this specific research and experimentation to be conducted by the Commission substantially as described in Application Exhibit XX. In addition, it will, of course, have available the extensive work in connection with the PRDC [fol. 4118] reactor to be done by APDA and organizations with which APDA has contracted. I am generally familiar with this APDA research program, since data so developed in connection with the PRDC reactor has been available to the Commission under a study agreement for about the last five years.

In my opinion, the programs to which I have referred in my testimony should make available the remaining technical information needed to determine definitively whether or not the PRDC reactor can be operated at its proposed Lagoona Beach site without undue risk to the public health and safety, and that, as indicated by the schedule I have

already outlined, sufficient information for this purpose should become available by the end of 1959, when construction of the PRDC reactor is scheduled to be completed. I am further of the opinion that the results of these programs will show either that the proposed PRDC reactor as now designed can be safely operated, or that design changes can be effected which will satisfy the necessary public safety standards.

[fol. 4119]

EXHIBIT 4

COMPARISON OF FAST REACTORS

	EBR-I	*EBR-II*	*PRDC*
POWER			
Thermal Mw	1.4	62.5	300x
Electrical Mw	0.2	20	100
Thermal Efficiency %	17	32	30
Max. Ht. Flux	209,000	1,100,000	1,095,000
Avg. Ht. Flux		750,000	765,000
Avg. Specific Power, Kw/Kg	18.1	300	670
Avg. Power Density Kw/L		1000	820
FUEL AND MATERIALS			
Crit. Mass Kg	52	210	450
% Enrichment	90	60	25
Amount of U-238, Tons	4	50	—
COOLANT AND STEAM DATA			
Na or NaK Inlet °F (Primary)	440	720	550
Na or NaK Outlet °F (Primary)	600	900	800
Na or NaK Inlet °F (Secondary)	420	580	500
Na or NaK Outlet °F (Secondary)	580	880	750
Steam pressure psig	400	1250	600
Steam temperature °F	527	850	720
NEUTRONICS			
Avg. Flux N/ cm 2/ sec	2.0×10^{14}	2×10^{15}	5×10^{15}
Avg. Core Energy, mev.	0.5	0.2	0.2

[fol. 4326] In evidence June 17, 1957, Tr. 3222

BEFORE THE ATOMIC ENERGY COMMISSION

Docket No. F-16

In the Matter of

POWER REACTOR DEVELOPMENT COMPANY

TESTIMONY OF ABEL WOLMAN
PROFESSOR OF SANITARY ENGINEERING
THE JOHNS HOPKINS UNIVERSITY

Background and Qualifications

My name is Abel Wolman. I am Professor of Sanitary Engineering at The Johns Hopkins University, Baltimore, Maryland.

I was born in Baltimore, Maryland, on June 10, 1892. I received the degrees of Bachelor of Arts in 1913 and Bachelor of Science in Engineering in 1915, and the honorary degree of Doctor of Engineering in 1937, all from The Johns Hopkins University.

I have been engaged since 1915 in the practice and teaching of the profession of Sanitary Engineering. During that period I have held the following public offices, among others, in the State of Maryland between the dates indicated:

Chief Engineer, Maryland State Department of Health, 1922-1939

Chairman, Water Resources Commission of Maryland, 1933-1940

Chairman, Maryland State Planning Commission, 1934-1945

I have served on numerous national and international committees in the field of Sanitary Engineering, including the following:

Chairman, Advisory Committee on Sanitary Engineering and Environment, Division of Medical Sciences, National Research Council, 1942 to date.

Chairman, Expert Committee on Environmental Sanitation, World Health Organization, 1949-1952.

[fol. 4327] Chairman, Permanent Sanitary Engineering Committee, Pan American Sanitary Bureau, 1942 to date.

Federal Member, Potomac River Commission, 1940-1950.

I have at various times served as a consultant to the U. S. Public Health Service, the Tennessee Valley Authority, the War Production Board, the U. S. Navy Department, the Surgeon General, the Department of the Air Force, the National Science Foundation, the Defense Department, numerous American cities and local government bodies, the Governments of the United Kingdom, Israel, Ceylon and Argentina, and a number of business enterprises. I am a Past President of the American Public Health Association and the American Water Works Association and a Past Editor of the American Journal of Public Health and the Journal of the American Water Works Association. I have been Professor of Sanitary Engineering at The Johns Hopkins University since 1937.

I have served as consultant to the Atomic Energy Commission on general wastes problems and particularly as Chairman until 1957 of its Stack Gas Working Group. I was a scientific advisor to the United States Delegation, International Conference on Peaceful Uses of Atomic Energy. I have served since 1947 as a member of the Reactor Safety Committee and the Advisory Committee on Reactor Safeguards.

I am generally familiar with the nature of the proposed PRDC reactor and the major safety problems associated with it, although I have not attempted to review in detail all of the technical information relating to the design of the reactor or to particular safety problems. While I did not attend the meeting of the Advisory Committee on Reactor Safeguards on June 3, 1956, I am familiar with the letter from Dr. McCullough to Mr. Fields of June 6, 1956,

and the recommendations of the Advisory Committee set [fol. 4328] forth in that letter. I have reviewed APDA-114, such portions of APDA-108 and Amendment VIII to the License Application as appeared to me to be relevant to environmental problems, the written testimony of Messrs. Bethe, Hilberry and Amorosi and such portions of their oral testimony as appeared to be relevant to environmental problems, the testimony of Dr. McCullough and Mr. Rogers, and the written testimony of Dr. Brooks.

The Function and Method of Approach of Sanitary Engineering

The sanitary engineer concerns himself with the impact of the environment upon the health and welfare of man. His functions historically extend over the adjustment of those features of the environment which may produce disease, nuisance or discomfort. In general, such areas of action relate to water supply, sewerage, milk and other foods, atmospheric pollution, industrial wastes, industrial hygiene, community refuse, etc.

The sanitary engineering discipline, therefore, requires a hybrid scientific underpinning consisting essentially of engineering training (in general, civil engineering) coupled with biological, chemical and physical sciences. The tools are composed of these combinations of the engineering arts and the basic sciences of biochemistry and biophysics.

In application, the sanitary engineer is dependent upon the industrial designer and operator for the disclosure of process, hazard to on-site personnel and surrounding public, nature and quantity of wastes and their relationship to the site as a whole. The sanitary engineer evaluates and appraises the impact on the surroundings of the industrial operations. In the handling of industrial wastes problems, for example, the sanitary engineer had to acquire a sufficient understanding of specific industrial processes [fol. 4329] to assess the origin, composition, volume and behavior of the resultant wastes. Since such wastes ultimately are returned to nature, via the air, the soil and the water, a large body of knowledge has been accumulated as to the behavior of such materials under the varying

natural conditions. Most of the orthodox industrial processes have become, in the course of time, sufficiently familiar to the sanitary engineer to permit him to make realistic appraisals of waste products and site problems. This does not mean that every sanitary engineer is a master of every industrial process, old or new. He is, however, either by personal experience or by literature search, provided with a reasonably detailed understanding of industrial processes, with particular reference to the behavior and the toxicity to humans and lower animals of any materials which may be released to the environment. Many quantitative criteria to aid in such judgments have been developed both theoretically and empirically.

The nuclear fission industry, new as it is, poses no strange or esoteric problems of site evaluation or of waste products management. The criteria as to toxicity and as to permissible limits for health and safet[illegible] new and still in evolution, but the principles and practices by which such criteria are applied to site evaluation and waste management are old and familiar.

General Criteria for Evaluation of a Site for Power Reactor

Site evaluation for any industrial facility requires, first, knowledge of the nature of the proposed facility and the kinds and quantities of possibly harmful materials which could be released from it both under normal operation [fol. 4330] and by an accident; and second, an understanding of the physical environment of the site insofar as that environment may have a bearing on the possible injurious effects to persons and property of any such release.

The basic criteria for evaluation of a site for a power reactor differ in no essential principles from those for the evaluation of a site for any other kind of industrial facility. A nuclear reactor has, however, special qualities, even when it can be classified as essentially stable and reliable in character. Its essential difference from other industrial facilities lies in the fact that even under normal operation a power reactor accumulates fission products in relatively large volume. These products are potentially more danger-

ous and more toxic than any other products known to industry. Indeed, Dr. McCullough has pointed out that fission products are more toxic per unit weight than any other industrially known materials by a factor of a million to a billion.

Under normal operation adequate provision must be made for the shielding of personnel from radiation released by these fission products and for the continuous or periodic safe withdrawal and disposal of these products. If an accident should occur, these accumulated products may be spread in one form or another within the operating area or, in some unfortunate instances, to areas very considerably beyond the confines of that area.

In site evaluation for a power reactor, therefore, four principal issues must be considered in the same manner as they would be considered in the case of any more familiar industrial hazard. They are:

(1) The types and quantities of release of harmful products that can occur from normal operation.

[fol. 4331] (2) The types of accidents that might occur and the probability of their occurrence.

(3) The nature and extent of property damage and personal injury that could result from either of the foregoing.

(4) The positive benefits to society, from the development of nuclear power, which must be balanced against these risks.

The Possible Hazards of a Reactor Accident

While careful attention must be given to the possible releases of toxic material from normal operation, the principal hazard with which one must be concerned at this stage of the development of nuclear power reactors is the possibility of a major reactor accident. This possibility, and the magnitude of its possible consequences, have been discussed in general terms in the excellent theoretical analysis, submitted by the Atomic Energy Commission to the Congressional Joint Committee on Atomic Energy on March 22, 1957, entitled "Theoretical Possibilities and Con-

sequences of Major Accidents in Large Nuclear Power Plants," and sometimes referred to as the "Brookhaven Report." In forwarding this report, the Acting Chairman of the Atomic Energy Commission stated:

> "The report serves to identify areas where continued research and development are needed, and areas where emphasis is needed in the further development of our regulatory program. It gives renewed emphasis to our belief that our research and development program in the nuclear power field must continue with vigor to the end that the 'conceivable' catastrophe shall never happen."

While I wholly agree with the necessity for doing everything possible to insure that a release of fission products outside the reactor site cannot occur, I do not believe that this report disposes in any way of the continuing necessity for careful analysis of the problems of site selection and [fol. 4332] evaluation for power reactors. As other witnesses have pointed out, the possible consequences of a bad reactor accident are so grave that one must rely on the cumulative safeguards of a series of protective barriers. One of the important protective barriers, in my judgment, is selection of a site such that, if other barriers fail, the adverse consequences of a release of fission products into the environment will be minimized. The importance of such siting considerations is of course heightened to the extent that the reliability of the instrument itself is subject to unresolved uncertainties.

It is undoubtedly true, as the Brookhaven Report points out, that major catastrophic reactor accidents are highly unlikely. The fact remains, however, that highly unlikely accidents have occurred in other industries. Familiar examples are the Texas City disaster on April 16, 1947, which resulted in over 570 lives lost, some 3500 injured, and property damage claims exceeding $50,000,000, and the Livonia fire at General Motors in 1953, which caused complete destruction of buildings and equipment at a loss of $50,000,000 and 2 deaths. The South Amboy, New Jersey, explosions of May 1950 illustrate a juxtaposition of im-

probable events. As a matter of fact, the events—which resulted in multiple explosions of adjacent plants, panic, high destruction of property, but fortunately little loss of life—might be considered a major comedy of errors of site selection, of fire fighting, etc. The significance of each of these incidents is that ~~they~~ did occur. Indeed, the conviction which grows upon me as the result of many years' consideration of problems of industrial hazards is that highly unlikely things do occur.

[fol. 4333] Serious accidents, whether industrial or disease epidemic, are often the result of a series of unexpected concomitant events—all in total of great improbability.

Comparison of the probabilities of such accidents as these with the theoretical probabilities of major accidents from nuclear power reactors is difficult if not impossible. In the nuclear field there has been a greater concern with safety than in any other industrial field; the attempt is to analyse and guard against possible accidents in advance and to provide multiple protective devices designed to fail safe. On the other hand, the nuclear technology has developed more rapidly than in almost any other industrial field and hence our operating experience with nuclear facilities, materials and phenomena is relatively small.

At the very least, there is no reason to suppose that the power reactor industry, unlike other industries, will be immune from major accidents. The fact that no lives have thus far been lost from operating reactors, while gratifying, rests upon too brief a period of experience to warrant any assurance that this state of affairs will continue. The fact is that accidents to nuclear facilities have not been rare. All that one can say is that the consequences of such accidents have not been serious in the sense that no lives have been lost, that the damages, although some times extensive, have been restricted to dollars, and that the sites of most such accidents have been relatively isolated from dense population centers.

The best known example is an accident to the NRX reactor at Chalk River, Ontario, which occurred on December 12, 1952, as the combined result of operators' errors, malfunction of equipment, and certain features of the

[fol. 4334] reactor itself. The accident melted and ruptured a number of fuel elements and caused radioactive water to leak out of the reactor; ~~it did not result~~ in any major release of fission products directly to the atmosphere. The dollar damages were great, over a year was lost in returning to operation, and the disposal of highly radioactive wastes—amounting to over a million gallons containing some ten thousand curies of long-life fission products—was and continues to be a major problem. The accident occurred shortly after the start of operations and hence at a time when the accumulation of fission products was at a minimum. It also occurred at an extremely isolated site. It would be most interesting to speculate how these quantities of waste material, or indeed the greater amounts of activity which could have occurred after a substantial period of operation, could be disposed of on a site near population centers. It is obvious that the cleanup problems following even such a non-violent accident could have very serious consequences on the safety of surrounding populations unless adequate facilities were available to dispose of large quantities of radioactive wastes.

A number of other accidents and incidents to nuclear facilities are collected in a publication of the Atomic Energy Commission entitled "A Summary of Accidents and Incidents Involving Radiation in Atomic Energy Activities," June 1945 through December 1955, TID-5360.

Reference has already been made in the testimony to the meltdown of EBR-I, the rupture of a fuel element in Clementine, and the destruction of Godiva.

The foregoing comments indicate that there remains an unfortunate gap between the mathematical predictability of accidents and their actual occurrence. This gap exists even in so-called conventional industries; it is presumably [fol. 4335] even greater in a novel industry whose technology and materials are not yet fully understood.

In recognition of this gap, it was for some time the policy of the Government to locate nuclear reactors at isolated sites. Beginning with the sodium intermediate reactor at West Milton, New York, it has become the practice in a number of instances, in recognition of the practical desir-

ability of a more accessible location, to rely for protection on a metal envelope as a substitute for distance. In my view such containment should be considered not as a substitute for the inherent safety of a reactor, but as merely an added factor of safety. Some may feel that any foreseeable reactor, once enclosed, should be and can be put anywhere in populated areas. My own judgment is against such a principle in the present state of understanding of the science and art of reactor design and operation. A container must provide for multiple openings. It must permit entrance and exit by operating, maintenance and technical workers. It must provide openings for utility servicing pipes, coolant systems, electrical connections, and the like. For example, Mr. Amorosi has testified, at page 2631 of the Transcript, that there are perhaps twenty large penetrations and several thousand small penetrations contemplated for the containment shell of the proposed PRDC reactor. Each perforation of the container, no matter how well done, creates opportunity for structural weakness, for unexpected opening, and for ultimate careless failures to close exits. Since personnel will be working inside the containers, some of these openings will presumably be opened and closed several times a day. Even though airlocks with double doors are provided, such locks do not provide absolute protection. Indeed, in underwater tunnelling fatalities have occurred because such airlocks [fol. 4336] were left open or failed. Should such an airlock happen to be open at a time when a serious reactor accident occurred, the protective value of the container could be seriously reduced or wiped out.

The Princeton Laboratory fire of only a few years ago is suggestive in this connection. A review of this fire disclosed an unfortunate series of failures of so-called safety mechanisms, some of them fantastically foolish in retrospect. The burning out of an electrical wiring installation early in the Princeton fire made quite worthless fire safety doors which could not be operated manually and therefore could not isolate a relatively small fire within a single room. The building was destroyed by the fire.

Proper containment is an important additional safeguard against the dispersion of toxic materials in case of an

accident. With proper design and operating procedures the chance of a failure of the containment structure can be minimized, but it cannot be wholly eliminated. In my judgment, therefore, containment is not a substitute for either a safe mechanism or a suitable site and should be used only in conjunction with a realistic appraisal of both the instrument to be enclosed and the location at which it is to be placed.

Looking at the questions of reactor hazards from my point of view as a sanitary engineer, the considerations which I have discussed above all serve to emphasize the importance of an adequate site evaluation. There must be a competent and thorough appraisal of the environmental hazards which might be occasioned either by normal operation or by an unexpected accident. The environmental conditions which should be reviewed include the geology, the hydrology, the meteorology, and the population distri-
[fol. 4337] bution in relation to the specific site or sites under consideration. In similar fashion, the community conditions surrounding the site must be reviewed with particular reference to the downstream users of ground and surface waters and the proximity of industrial and residential installations, agricultural activities, and recreational areas. Their distance, their nature, and the possibility of interference with their use become issues in appraisal. Review must be made of the characteristics of gaseous, liquid and solid wastes which can result from both normal operation and accidental conditions; their quantity and quality must be considered, the treatment facilities to be provided, the adequacy of storage and the nature of ultimate disposal facilities must be reviewed, all from the point of view of assuring that adequate means are available to prevent a release of toxic materials to the surrounding environment. Since considerable use is made of nature itself in the evaluation of dilution or concentration factors, careful investigation must be made of all natural features which will tend to dissipate or increase the hazards.

It would resolve a great many difficulties and debates if all of the elements which go into the appraisal of a site could be translated into a simple mathematical formula.

Unfortunately, this is not possible. The number of parameters involved in such a formula is great and they do not lend themselves to mathematical statement or accuracy. Continuing efforts have been made to adapt these qualitative appraisals or judgments to a mathematical formula. Generally, these formulae have sought to express desirable exclusion distances as a function of magnitude of power to be developed. Both the validity and usefulness of such formulae appear doubtful. It is probable that site selection for power reactors will continue for many years to [fol. 4338] come to be a field for detailed appraisal by qualitative judgments, tempered by experience, accident occurrence, and philosophy of regulation. To permit such an appraisal, the most complete information should be available for any site which is to be considered for a reactor of significant power. Estimates must accurately be made not only of the probability of accident, but of the measures which would be taken to avoid risks to employees and to offsite populations in managing the consequences of such an accident.

In general, it is apt to be true that the proponents of a particular reactor show the highest degree of optimism regarding both the machine and its site, while those most directly concerned with the protection of the health and safety of the public demonstrate a less degree of optimism. This juxtaposition of judgments is neither unfamiliar nor unexpected. Historically, it accounts for the fact that site locations for industries of more familiar and normal characteristics usually have external review by public regulatory agencies. This situation has arisen not so much because of any philosophical difference of opinion, but because experience has shown that industry is not always the best judge of the disability which it might create for the public. Examples of invalid assumptions in this respect by conventional industries could be listed by the dozens.

Environmental Problems of the Location of the PRDC Reactor

In applying the general principles discussed above to evaluation of the environmental problems presented by the

PRDC reactor and its proposed site, one must start with an evaluation of the accident hazards presented by the PRDC reactor itself. I do not, of course, purport to be [fol. 4339] expert on the questions of nuclear physics and engineering which are predominant in a determination of these accident potentials. From such portions of the License Application and the record in this case as I have read, however, it would appear that the safety and reliability of the reactor are not yet established, but remain to be established by additional research and development work to be performed prior to the commencement of its operation. If that work establishes that the reactor is inherently safe and reliable, and that no credible accident can release fission products into the atmosphere, then the proposed site would be satisfactory from the point of view of its accident potentiality. This, however, could only be determined on the basis of a more definitive analysis of the hazards potential of the reactor than I have yet seen. If, on the other hand, the research and development work is such as to leave substantial remaining doubts as to the safety and reliability of the reactor, and the possibility that an accident could release fission products into the atmosphere, then the suitability of a location in a populous area and within 30 miles of two major cities is open to very serious question.

As I have stated, in reaching a judgment on siting, I must rely on the judgments expressed by nuclear physicists and nuclear engineers to determine the accident potential of the machine. From the testimony of Hilberry, Bethe, McCullough, Rogers, and Brooks, it is quite apparent to me that there exists as of the present date a disagreement among the experts concerning the safety of the type of reactor proposed by PRDC. On the one hand, Hilberry and Bethe seem satisfied based on present experimental information and certain theoretical calculations. On the other hand, both Rogers and McCullough have stated ex- [fol. 4340] plicitly that they still agree with the conclusions set forth in the June 6, 1956, letter signed by Dr. McCullough on behalf of the Advisory Committee on Reactor Safeguards, while Brooks states that "I do not believe

that the presently existing supporting data are sufficient to give us the kind of confidence we ought to have to permit a project of this magnitude to be operated."

In short, the present views of those qualified to have opinions on the questions of nuclear physics and nuclear engineering involved cover a fairly wide spectrum, ranging from fairly complete assurance of safety to serious doubts which cannot be resolved for several years. Faced with such a divergence of views I, as a sanitary engineer engaged in evaluating a site, can only conclude that the suitability of the proposed site has not been established. In particular, I cannot, in evaluating the site, accept only the optimistic views of Bethe and Hilberry and disregard the views of other competent men who believe that the inherent safety of this type of machine has not been and probably cannot for some time be, established.

Unless and until the inherent safety of the reactor, and the ability of its various structures to contain a serious accident, are established to the satisfaction of most of the people competent in these areas, it is evident that very serious questions are presented from the point of view of sanitary engineering as to the suitability for such a reactor of the proposed site, or indeed any site in relatively close proximity to large centers of population. These questions may, of course, become resolved by the results of work yet to be done, but this does not alter my conclusion based on the present state of knowledge.

In addition to these unresolved questions of accident potential, there is also a serious question as to the capacity of the proposed site to handle the problems of waste [fol. 4341] disposal which could arise in the event of a serious contained accident. Mr. Amorosi has testified that "very little study to date" has been made of the problem of cleanup of a major incident. (Transcript p. 2590; see also p. 2592.) Hence, it is impossible for me to say at the present time whether it will be possible at the proposed site, by provision of adequate facilities and procedures, to deal adequately with the cleanup problems which could arise, or whether there may be features of the site which seriously increase the possible public hazards attendant on such cleanup.

In this connection, I point out that the tenor of Mr. Amorosi's testimony, as I understand it, is that it is not necessary at this time to have developed the procedures and methods of handling the wastes which might result from a contained accident because he feels confident that it will be possible safely to put those wastes into containers and transport them off site. In my judgment, the quantities of radioactive materials which could result from a serious contained accident are sufficiently large and the hazards attendant upon release to the environment of even a part of those materials sufficiently great so that one cannot be sure that adequate means of disposal of those wastes at the proposed site will be available unless the problem has been carefully studied. When there is a contained accident, one can do any of several things. One can shut the facility down and permit the fission products to remain inside for an indefinite period. In this event there are risks of leakage or seepage through the container, the structures, and the soil which would have to be analysed; the extent and seriousness of these risks may be dependent in part on meteorology, hydrology and ground seepage. [fol. 4342] Alternatively, one can remove the material into on-site disposal facilities; here again, the meteorology, hydrology and soil conditions are highly important. Finally, one can enclose the material in containers and transport it off-site to a safe storage area. The transportation of large quantities of radioactive wastes through a populous area involves other hazards of a possibly serious nature. Moreover, some or all of these courses of action may entail costs which are so great as to approach being prohibitive. Until the problem has been studied and general plans for the handling of contamination resulting from a contained accident have been developed, I do not feel that I am in a position to form any conclusion as to the practicability of dealing with such problems at the proposed site.

The environmental information which has been submitted in connection with this application, and which I have attempted to review in detail, does not afford an adequate basis for an evaluation of the particular site. The information is contained primarily in the monograph entitled APDA-114. This document may be fairly classified as a

purely introductory evaluation of environmental problems created by this site. It leaves unanswered a variety of questions upon which any decision as to the validity of the site could be made. It is studded with recommendations for future very much more detailed studies. In virtually no instance do the authors of the various portions of the document attempt to give their conclusions even the semblance of definitiveness.

For example, it appears that surrounding densities of population have been analysed only for a ten-mile radius (Appendix VII, p. 2). The meteorological studies (Appendix IV), although described as a final report, state that they are "hampered by the absence of knowledge of [fol. 4343] the diffusion pattern over and near a lake surface" (p. 29), and recommend a program of meteorological observations at or near the plant site, and of studies of diffusion over the lake under various meteorological conditions (p. 20). The results of neither of these are available. Mr. Amorosi's testimony indicates that a steel tower has been erected at the plant site and that observations are being taken from it but have not been analysed, while the diffusion studies have not commenced (Tr. 2584). Information as to quantities of radioactive wastes from normal operation is not yet available (Tr. 2587), and very little study has been made to date about the problems of cleanup of a major incident (Tr. 2590-92). Apparently little or no study has been given to problems of site inundation. Studies of the possibility that the lake level might exceed presently recorded high water marks are incomplete (Tr. 2583). Water and air samples are still to be obtained (Tr. 2580). I find little or no discussion of the effect of an accident at the site upon water intakes north and south of the site area. Information on sources of potable and industrial water supply is still to be furnished (Tr. 2582). In general, the testimony indicates that essentially the only area of environmental information which even Applicant regards as complete is the seismological data (Tr. 2597).

Such information as has been submitted discloses the following specific problems with respect to the proposed

site, which in my judgment would have to be resolved before a determination could be made as to the suitability of the site for a power reactor whose stability and basic safety had been established:

(a) The meteorology of the site is quite complex because of its proximity to the lake. The high frequency of fogs of long duration at certain seasons of the year, other natural manifestations of inversions and highly erratic air [fol. 4344] currents are commented upon in APDA-114. Their detailed identification still remains undocumented, and in particular there is inadequate discussion of the behavior of gaseous and fine particle releases into the air. In my judgment, probably few sites in this region would pose as many meteorological issues as the one here chosen.

(b) Since, as I understand it, chemical processing of fuel will not be carried out at this site, there is a natural concern as to the manner in which spent fuel elements will be transported off-site and any possible hazards attendant upon such transportation. I have seen no discussion of this problem in the evidence.

(c) The location of municipal and industrial water intakes with respect to the site has not been fully indicated. The effect of discharges (by accident or by intent) upon such intakes should be evaluated, with particular reference to water currents, wind action, etc.

(d) Additional study appears necessary of the possibility of flooding of the site and the extent to which such flooding could result in premature dispersal of radioactive wastes in the hold-up tanks or lagoons.

Based on the information available, it is my judgment that with adequate containment and assuming the construction of adequate facilities and adoption of adequate procedures for the disposal of normal operating wastes, surrounding populations and properties would probably not be endangered by normal operation. The information submitted affords an inadequate basis, however, upon which to rest any evaluation as to the environmental hazards which would be presented either by severe floods under normal reactor operation or by the cleanup of a contained

[fol. 4345] accident. Finally, neither the technical nor the environmental information affords an adequate basis for the evaluation of the hazards to the environment which would be created by an uncontained accident. In my judgment, therefore, any definitive determination with respect to the suitability of the proposed site for the proposed reactor cannot be made until substantial additional information has been supplied.

[fol. 369] Cross-examination of Ernest R. Acker by Mr. Sigal, Attorney for Intervenors, on March 4 and March 5, 1957, Tr. 369-70, 372-73, 467.

* * * * * * *

Q. In the light of the fact that PRDC says it will not under any circumstances operate this reactor unless it is safe, why wouldn't $65 million be adequate coverage?

A. Mr. Sigal, I assume if you were advising a client, even if they had what they considered to be and maybe what you considered to be a completely safe operation, which we think we can ultimately achieve, I think you as a lawyer would advise them to take whatever insurance they could get or whatever they felt was necessary to meet the remotest ultimate risk and that is our philosophy with respect to insurance.

Q. Insurance is not based on the determination of what the remotest risk is, Mr. Acker. Isn't it based on what experience shows to be the risk in given situations?

A. I would say that the rates are based on the experience with respect to various classes of risk, yes. But I don't know of any experience background on which that could be determined in this case.

Q. So in this situation all those who are connected with this problem are unable to put any limit on the possible liability that may flow from the operation of a reactor? Is that correct?

A. I don't know as to that, Mr. Sigal. I am not willing to admit that freely.

Q. PRDC takes the position that is it not possible now to put any top figure less than $500 million on the possible

liability arising from the operation of this plant, is that right?

A. I would say that is a fair statement.

* * * * * * *

Q. Mr. Acker, you stated in your testimony on direct examination earlier that the companies which are members of PRDC have assets of over $6 billion and an equity of over $2 billion.

A. That is right.

Q. Why wouldn't those companies underwrite the risk in operation of this plant if the risk is no more than $500 million?

A. I think for the very same reason that very few companies are self-insurers.

[fol. 370] Q. If you can't get insurance from private companies, then you would be obliged to do self-insurance.

A. We would have to consider whether we wished to risk the assets of our stockholders to provide for that kind of contingency.

Q. You have not considered that so far?

A. No.

* * * * * * *

Q. As a matter of fact, I assume you know, Mr. Acker, that no fast breeder reactor has been built which has survived all tests made?

A. I think that is a fact.

* * * * * * *

[fol. 4858]

Commission Exhibit No. 3

(The letter of transmittal accompanying the following Report is printed at pp. 349-356 of Volume II of the Record.)

United States
Atomic Energy Commission

THEORETICAL POSSIBILITIES AND CONSEQUENCES OF MAJOR ACCIDENTS IN LARGE NUCLEAR POWER PLANTS

A Study of Possible Consequences if Certain Assumed Accidents, Theoretically Possible but Highly Improbable, Were to Occur in Large Nuclear Power Plants

March, 1957

[fol. 4859] United States

Atomic Energy Commission

This report to the Commission contains an account of a study undertaken by the Division of Civilian Application, at the direction of the General Manager, to gain a more comprehensive understanding of the potential public hazards of nuclear power reactors.

All technical phases of the project were performed by a study team composed of staff members of the Brookhaven National Laboratory, with assistance of consultants and others from elsewhere. Principal contributors were:

Dr. Clifford K. Beck
Dr. Frederick P. Cowan
Mr. Kenneth W. Downes, Project Director
Dr. Joseph A. Fleck, Jr.
Dr. J. B. H. Kuper
Mr. James McLaughlin
Mr. Irving Singer
Mr. Maynard Smith

The study was carried out under the guidance of a Steering Committee composed of scientists and engineers of the

Atomic Energy Commission staff and the Brookhaven National Laboratory. Members were:

Dr. Clifford K. Beck, AEC, Chairman, Steering Committee
Dr. Walter D. Claus, AEC
Mr. Kenneth W. Downes, BNL
Mr. Merrill Eisenbud, NYOO
Dr. Clark Goodman (replaced by Mr. Howard Hembree, AEC)
Mr. Edwin A. Lamke, AEC, Secretary
Dr. Gerald Tape, BNL
Dr. Clarke Williams, BNL

Valuable assistance throughout the study was also rendered by Mr. Joshua Z. Holland, AEC, and in some of the technical phases by Mr. Raymond O. Brittan, Argonne National Laboratory, and Mr. Everitt P. Blizard, Oak Ridge National Laboratory.

Many other staff members, consultants and advisers, including, members of the Advisory Committee on Reactor Safeguards also rendered valuable assistance in the study.

[fol. 4860] Table of Contents

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[fol. 4861] INTRODUCTION

[fol. 4862] INTRODUCTION

It might be supposed, because the essential fuel in a nuclear power reactor is the same as that in atomic bombs, that gross malfunctioning in power reactors could possibly lead to a devastating explosion similar to those produced by A-bombs. Such is not the case. Under no conceivable circumstances can accidental nuclear explosions in power reactors cause significant direct public damage beyond the boundaries of the exclusion areas around such installations.

There could be explosive nuclear energy releases in power reactors, or chemical or physical energy releases from components or auxiliary systems, sufficient in magnitude to destroy the reactor, possibly break the various containment structures within which it is housed and wreck the auxiliary machinery. Such an accident would constitute a real threat to the life of personnel within the facility and could result in complete loss of the expensive installation. Nevertheless, little hazard to the general public would ensue from the explosion itself.

There is, however, another hazard to the general public which could cause extensive loss of life and damage to property. This is the possibility of radiation exposure and [fol. 4863] contamination, if the fission products stored up in the reactor should be released. It is possible to conceive of accidents which would release the accumulated fission products from a large nuclear reactor in a finely divided state so that a significant portion of them would become airborne and subject to atmospheric dispersal over wide areas. Injury or death could result to people from exposure to the direct radiation from these materials, or from ingestion of portions into the body. Settling out of these materials could cause both further hazard to health and costly contamination damage to property. Death at distances of many miles and injury and property damage for hundreds of miles could conceivably occur.

Fortunately, radiation intensity from most fission products released from a reactor decreases rapidly. The possibility of total release is exceedingly remote, and among those products most likely to be released are those which decay most rapidly. In no conceivable way could fission products from a reactor be distributed rapidly and uniformly over large areas. The major threat to the safety of people remote from the site of release would not be instantaneous; periods up to hours and even days after release would be available within which to avoid the full effects of radioactivity from the fission products.

It must be clearly recognized, however, that major releases of fission products from a nuclear power reactor conceivably could occur and that a serious threat to the health and safety of people over large areas could ensue.

[fol. 4864] An over-all appraisal of the actual magnitude of hazard to the public arising from operation of a nuclear power reactor revolves around the best possible answers to four essential and difficult questions:

1. What is the likelihood that fission products might be released?

2. What are the factors and conditions which would affect the distribution of released materials over public areas?

3. What are the levels of exposure or contamination which cause injury to people or damage to property?
4. If releases of fission products should occur, what deaths or injuries to people and costs in damaged property could ensue?

Succeeding sections of this report are devoted to consideration of these questions.

It is important to recognize that the magnitudes of many of the crucial factors in this study are not quantitatively established, either by theoretical and experimental data or adequate experience. Appraisal must rest on the judgment and considered opinions of the most knowledgeable persons in the field. At various places in the report note will be made where important components are particularly uncertain, but it must be remembered continuously that this entire study hardly constitutes more than an identification of the factors which are important, the best appraisal of these [fol. 4865] factors currently possible, and a rough approximation of the magnitudes of the composite results.

There are many essential and significant qualifications and uncertainties in the conclusions contained in this report. If separated from these qualifications and uncertainties, the conclusions would lose their validity. However, we believe that the study, if taken in perspective, gives an order-of-magnitude frame of reference, and defines tentative boundaries for this problem.

More definitive information on estimated factors would probably tend to reduce the estimates of damages, though in a few instances the converse might be true. There are a few less usual weather conditions which occur perhaps 5 percent of the time and which could yield estimated damages outside the range of the figures stated here. Therefore, this study does not set an upper limit for the potential damages; there ~~is no~~ known way at present to do this. It does indicate the range of hazards from highly improbable catastrophic reactor accidents which might occur under all except a small percentage of most unusual combinations of circumstances.

[fol. 4866] PART 1

THE PROBABILITY OF CATASTROPHIC REACTOR ACCIDENTS

THE PROBABILITY OF CATASTROPHIC REACTOR ACCIDENTS

[fol. 4867]

The probability of occurrence of publicly hazardous accidents in nuclear power reactor plants is exceedingly low.

This single statement, re-emphasized, would suffice to report this portion of the study, except for the essential importance of this central fact of "low probability" to comprehension of the over-all public hazard of power reactors. The significance of damages consequent to accidents cannot be appraised independently of the probability of the accidents.

One fact must be stated at the outset: no one knows now or will ever know the exact magnitude of this low probability of a publicly hazardous reactor accident. In trying to establish some estimation of this quantity, three possible approaches might be used:

1. Operate enough reactors for sufficient length of time to obtain an indication of the accident probability.

2. Give careful consideration and approximate numerical values to all separate factors which would either prevent or cause such an accident, then try to calculate, or guess, the composite result of these factors and hence the likelihood of occurrence of accidents.

[fol. 4868] 3. Obtain a weighted average of the best judgments and judicious opinions of the most experienced and knowledgeable experts in the field.

None of these approaches is satisfactory. Even when combined, they are at the present time, still unsatisfactory.

Indications from Cumulative Experience to Date

Nuclear reactors have been operated since December 2, 1942, with a remarkable safety record. We have accumu-

lated more than 100 reactor years of experience with large routinely operated reactors without any accidents.* This record of safety, although highly reassuring, does not afford a dependable statistical basis for estimating the probability of occurrence of serious reactor accidents in the future.

In this initial period of power reactor experience, types of reactors, detailed reactor designs, and operating patterns are all experimental and variable.

There are factors both on the side which would lead toward confidence that our "accident" experience will continue, and on the converse side. On the one hand, we [fol. 4869] attempt to provide wide margins of safety because of our limited knowledge of accident potentials of reactors. The new and glamorous field challenges and attracts the most expert and competent people. The Government has had and continues to have a substantial safety research program. Experience almost certainly will lead to safer design. On the other hand, since many reactor types are being developed more varied safety problems may exist than would be the case in fewer types. Accident free experience could lead to complacency. Lengthening reactor life could lead to hazards not otherwise encountered (cumulative radiation damage to components). Competitive pressures could furnish incentives to reduce margins of safety.

Factors For and Against a Major Accident

It is very difficult to determine whether a reactor of one type is safer, overall, than one of another type. It is easy to point out superior safety features and inferior ones in any one type compared with those in another type. Safety depends on the combination of many complex and inter-related factors and overall comparison of one reactor

* All the half-dozen "runaway" incidents (Chalk River, Borax, EBR-1, etc.) experienced thus far, either inadvertent or planned, have occurred in research or experimental test reactors—in contrast to the steadily operating power reactors considered here. No one has been injured, and no fission products have been released "off-area". Hence, the accidents are not in the category of concern in this study.

type with another depends on value judgments which are difficult to define quantitatively.

To estimate the *absolute* safety of a given reactor, or of reactors in general, or to estimate the quantitative, probability that an accident will occur is more difficult, and more uncertain, by several orders of magnitude, than is the relative comparison of reactors.

[fol. 4870] In principle, it should be possible to identify each factor, positive or negative, involved in the safety of a reactor, assign some measure of the magnitude of its effect and some probability of its functioning (or failing to function), then derive a net weighted composite measure of the margin of safety, or of the probability of catastrophic accident in a given time.

On the positive side would be such factors as:

1. In no reactor, so far as is known, will a single equipment failure or a single operating error lead to a fission product-releasing accident (even within the containment structure). If such condition were recognized, it would be rectified. In the vast majority of cases, multiple separate malfunctioning events are a necessary prerequisite to a serious accident.

2. Most reactors are inherently stable, e. g., most reactors possess prompt negative temperature or power coefficients (any increase in these factors is accompanied by a decrease in reactivity, hence, any excursion tends to reach some limiting value, rather than indefinitely increasing power).

3. In heterogeneous (solid fuel) reactors, the fission product inventory accumulates within the solid fuel matrix from which escape is prevented not only by low mobility of these fission products in the solid fuel but also by the metallic surface cladding. Melting or violent damage must occur before fission products can be released into the reactor vessel. In homogeneous (solution or slurry fuel) reactors, the possibility of continuous removal of the fission products offers some compensation for the lack of confinement provided within the fuel elements of other types.

[fol. 4871]

4. Every power reactor will be provided with an adequate primary containment vessel enclosing the reactor core within which fuel and fission products reside. This, in turn, is surrounded by massive radiation shields for biological protection of workers.

5. All power reactors now considered for construction in populated areas are provided with "vapor shells" designed to contain all fission products that might be released in any credible accident.

6. Seventy-five or eighty percent of the fission product elements are solids at ordinary temperatures and, unless opening of the outer vapor shell is caused or accompanied by an event which vaporizes and violently disassambles the core materials, most of the fission products would be expected to remain attached to fragments of fuel elements or to settle out on nearby structures.

7. Should fission products be released from the containment shell, not only the physical state of the materials, but also a complex variety of environmental meteorological and other factors, having various probabilities of occurrence, would govern the subsequent pattern of dispersal. Probabilities of progressively unfavorable combinations of conditions become progressively lower, so that likelihood of highly unfavorable combinations is extremely low.

On the negative side, account would have to be taken of such factors as:

1. Many power reactor systems will operate under high pressures. High pressure systems are subject to failure.

2. The cumulative effect of radiation on physical and chemical properties of materials, after long periods of time, is largely unknown. Eventual serious failures may occur.

3. Various metals used in reactors such as uranium, aluminum, zirconium, sodium and beryllium, under

certain conditions not at present clearly understood, may react explosively with water, also present in many reactors. During incidents of abnormal opera- [fol. 4872] tion resulting perhaps in melting of some of the metals in contact with water and under the influence of radiation, chemical reactions of enough violence to rupture the containment vessels, with release of the fission products, could occur.

4. After initial operation, many of the vital components become inaccessible for inspections. In non-nuclear plants, serious accidents are often averted through detection of incipient failure.

5. Much remains to be learned about the characteristics and behavior of nuclear systems.

Listing of such items, in both positive and negative tabulations, could proceed at length. However, it should be clear already that, even if all the significant factors relevant to safety were known, it would be essentially impossible to assign dependable quantitative values to their respective probabilities of functioning and to derive therefrom a reliable indication of the margin of safety under operating conditions likely to exist.

The Best Judgment of the Most Knowledgeable Experts

Many outstanding leaders in reactor technology and associated fields were consulted in the course of this study. It is their unanimous opinion that the likelihood of a major reactor accident is low. There is a general reluctance to make quantitative estimates of how low the probability is. There is a common aversion to attachment of quantitative estimates to a phenomenon so vague and uncertain as the probability of occurrence of catastrophic accidents, particularly since such assignment of numerical estimations [fol. 4873] conveys an erroneous impression of the confidence or firmness of the knowledge constituting the basis for the estimate. Also, some hold a philosophic view that there is no such thing as a numerical value for the probability of occurrence of a catastrophic accident; that such a thing is unknowable.

Thus, many decline to make even order-of-magnitude guesses of the probability of catastrophic reactor accidents. On the other hand, a few have ventured to express their confidence of the extremely low probabilities of occurrence of such accidents by stating numerical, order-of-magnitude estimations. An indication of the range of these is illuminating.

Should some unfortunate sequence of failures lead to destruction of the reactor core with attendant release of the fission product inventory within the reactor vessel, however expensive this would be to the owners, no hazard to the safety of the public would occur unless two additional lines of defense were also breached: (1) the integrity of the reactor vessel; and, (2) the integrity of the reactor container or vapor shell.

Accidents of sufficient violence to breach these successive lines of defense occurring concurrently with progressively unfavorable combinations of dispersive weather conditions have decreasing probabilities of occurrence.

Thus, the probability of public hazards from reactor [fol. 4874] accidents may be considered in terms of a sequence of events, each being prerequisite to the situation arising from succeeding events, and each having a lower probability of occurrence than its predecessor. As indicated above, the numerical estimates ventured here represent an attempt to express in numerical terms the degree of feeling held by some of our advisers for the remoteness of the possibilities of occurrence of the various accidents described. It should be emphasized that these numbers have no demonstrable basis in fact and have no validity of application beyond a reflection of the degree of their confidence in the low likelihood of occurrence of such reactor accidents.

Their estimates for the likelihood of destruction or major damage to the reactor core with significant internal release of fission products, but no release outside the reactor vessel, ranged from one chance in 100 to one in 10,000 per year for each reactor.

Their estimates for the likelihood of accidents which would release significant amounts of fission products outside the reactor vessel but not outside the containment

building (the contained accident) ranged from one chance in 1,000 to one in 10,000 per year for each reactor.

Finally, their estimates for the likelihood of accidents which would release major amounts of fission products outside the containment (the major release accident) ranged from one chance in 100,000 to one in a billion per year for each reactor.

[fol. 4875] Taking the most pessimistic of these estimates for the major accident, assuming that 100 reactors are in operation in the United States, and making the unrealistic assumption that each accident of the type defined would kill 3,000 people, there would be one chance in 50 million per year that a person would be killed by reactor accidents. For comparison, the chance of a person in the United States being killed by automobile accidents, assuming that each person has an equal likelihood of being among the 40,000 killed, is about one in 5,000 per year.

Safety Through Safeguards

Detailed evaluation of the safety of a reactor before approval is given for its operation may not lead to any better estimations of accident probabilities than those yielded by other considerations, but it does furnish added confidence that accident probabilities are indeed exceedingly small. In fact, the confidence of many persons in the low probability of accidents is due in large part to the application of these evaluation procedures.

Three aspects of these procedures contributing to minimization of public hazards from reactor accidents are worthy of mention:

1. Knowledge that safety evaluations and reviews are prerequisite to operation approval insures attention to and emphasis on safety aspects of a facility at all stages of the design.

[fol. 4876]

2. The detailed safety analysis and evaluation by experts on the Commission staff, with assistance as necessary from consultants and advisers, including the Advisory Committee on Reactor Safeguards, assures that at

least one independent review is given to each reactor facility in addition to that given by the designers.

3. As a part of the pre-evaluation procedure, careful analysis must be given to establishment of the accident of maximum proportions considered to be credible for each reactor facility, and demonstration must be made that adequate safeguards are provided the public against this eventuality.

Thus, since there is protection against "credible" accidents, no damages to the public will occur unless "incredible" accidents take place. It must be recognized, of course, that errors in judgment can be committed, with resulting occurrence of what was believed to be an "incredible" accident. Nevertheless, the consistent and rigorous execution of these procedures for every reactor warrants a considerable degree of confidence that safeguards against serious accidents have been incorporated, and that the probabilities of such occurrences are small.

[fol. 4877] PART II

ASSUMPTIONS USED IN THE DAMAGE STUDIES

[fol. 4878] ASSUMPTIONS USED IN THE DAMAGE STUDIES

It has been concluded that there is some remote but quantitatively uncertain possibility that a major reactor accident might occur. The immediate question then follows: What could be the extent of consequent damages? The remaining sections of this report devote attention to this question. Consideration is restricted to estimation of the damages to the public. No attempt has been made to appraise the hazard or damage to the facility itself or to its personnel.

To evaluate the hazardous consequences to the public of a reactor accident of major proportions, many features must be further described relating to the size and location of the reactor, its fission product inventory and the portion released, the conditions of release and the features of its delivery to public areas. In this section of the report,

brief definitions and descriptions of those situations and features considered pertinent are recorded. Details of the technical foundations for these assumptions and specifications, and mathematical manipulations to arrive at estimates of the consequences thereof, are contained in various appendices as indicated.

Two comments are appropriate at this point. 1) Conditions and specifications described below are chosen to be representative of a "generalized" power reactor situation. [fol. 4879] Specific reactor situations will vary somewhat from the one described herein; however, use of the generalized reactor and site is adequate to permit a reasonable evaluation of general public liabilities. 2) The assumptions and specifications are chosen to be on the pessimistic side, i.e., result in higher damage estimates. This is due to an attempt to be on the safe side where uncertainties exist in present knowledge but no deliberate safety factors have been introduced.

Typical Reactor

The reactor considered is of 500,000 Kw thermal (100,000-200,000 Kw electrical) steadily operating, power producing type, having an average fuel reloading (and fission product eliminating) cycle of 180 days. Accidents assumed in this study, described later, are postulated to have occurred near the end of the 180 day cycle, when fission product inventory would be maximum. Research and test reactors and reactor experiments are excluded from consideration. A leak and pressure-resistant containment building of the usual type is assumed to surround the reactor.

Fission Product Content of the Reactor

For the 500,000 Kw thermal reactor, 180 days of operation, the fission product inventory would be approximately 4x10' curies, when measured 24 hours after an accident (or shutdown). Decay of the fission products as well as their composition was taken into consideration for calculation of direct radiation exposures or contamination due to deposition. Special attention was given to the volatile [fol. 4880] fission products, xenon, krypton, bromine and

iodine and to strontium. The latter two are biologically the most hazardous.

Typical Location

The reactor is assumed to be located near a large body of water, most likely a river, and about 30 miles from a major city. As in many sites proposed to date, a site boundary of 2000 feet radius is postulated.

Population Distribution

Distributions of populations around reactors would differ considerably in detail from one site to the next. However, many general features would be remarkably similar, especially at large distances. Each reactor site would be in an area of low population density, a large city would be located about thirty miles away and the density of population would increase from the reactor toward the city. If the total population enclosed within a circle of radius R centered at the reactor, is calculated for distances to the city, it develops that the total population within given radial distances is remarkably alike for all reactor sites now in use or proposed. This population can be expressed by the equation: $P = 200R^{2.83}$, where R is in miles.

At distances beyond the city, the average population density decreases and a different expression must be used. Average population density over the entire United States [fol. 4881] is about 55 people per square mile. Reactors, however, are likely to be built in more populated areas such as in the northeast where the average runs about 500 per square mile. Therefore, the assumption is made that the population density beyond the city is constant, and averages 500 people per square mile. For most situations these assumed population distributions are on the conservative side, i.e. in hardly any likely place would the population be underestimated, and in most places they overestimate the number of persons in areas which may be affected by a reactor accident.

For some types of accidents, the high population density in the nearby city needs to be calculated independently of the general treatment described above. In these cases, it

was assumed that the city located 30 miles from the reactor has a population of about 1,000,000 persons spread uniformly over a region having a diameter of about 10 miles. Where the existence of the city contributes significantly to the calculated damages, city damages are listed separately.

Characteristics of Released Products

Accidents of greatest concern would be those which resulted in release and subsequent atmospheric dispersal of fission products from the reactor. The characteristics of the fission products at the time of release would have a great influence on their subsequent dispersal. Two factors having the greatest impact in determining the effect of distribution due to various meteorological conditions would be the size of the particles contained in the release and the [fol. 4882] temperature of the radioactive cloud at the time of release. These factors could, of course, vary from one reactor accident to another and undoubtedly would be highly dependent upon the particular accident. For the purpose of this study two choices were made for each factor, each choice being considered as probable and also illustrative of widely different conditions. For temperatures of release, the two *chosen* conditions were characterized by "hot" and "cold", the temperatures being 300° F (temperature of steam at a pressure sufficient to rupture the containment vessel) and 70° F (normal atmosphere temperature), respectively. For particle size two distributions were assumed, one centered about one mic on and the other seven microns in diameter, these being re esentative of fumes and industrial dust, respectively. Experience does not permit a better definition of the particle size; it does, however, lend credibility to these two choices.

Mechanism of Distribution

Assuming that a release had occurred, consideration must then be given to the assumed existing weather conditions and to other factors that might influence the rate and pattern of distribution of the released materials. Numerous variables here could combine into an almost infinite variety

of situations. It is possible (see Appendix I) to obtain an indication of the range of damages from calculations on a reasonably small number of cases by limiting the number of meteorological variables to those having major influences and choosing one or two appropriate values for each.

[fol. 4883] The meteorological variables selected are: weather—(a) dry and (b) rain (.02 inches per hour over the whole area affected); atmospheric stability—(a) typical daytime lapse with a wind speed of 5 m/sec (12 mph) and (b) nighttime typical inversion with a wind speed of 3 m/sec (7 mph) up to 50 meters height and 15 m/sec (35 mph) above; height of cloud rise—(a) cold release, zero, (b) hot release, 860 meters during lapse, 400 meters during inversion. (Appendix E). It should be noted that the conditions assumed in any given case existed continuously for the duration of the case and the area affected.

It should be noted here that exceedingly little is known about the details of atmospheric distribution, even if the characteristics of the materials under consideration and the many environmental factors involved could be stated with great confidence. Nevertheless, use of these approximate average values, above, does give reasonably dependable general indications of the results to be expected in a large majority of possible situations.

Tolerance Levels for Personal Injury

Personal injury might result from exposure of personnel to the radioactive cloud released during the postulated accidents. Personal injury might also arise from exposure to deposited fission products. In the latter case, ample time often would be available to permit evacuation from contaminated areas before serious injury would occur. In appraising the hazard to individuals who might be exposed, [fol. 4884] it would be necessary to define the probable extent of injury caused by various doses of radiation. This is an exceedingly complex matter (Appendix D). Using the best advice available and considering various biological effects such as ingestion, external and internal radiation problems, and the special problems arising from particular

fission product isotopes having special biological importance, the following ranges, as described in Appendix D, were adopted:

	Equivalent Whole Body Gamma Radiation	Concentration of Released Fission Products to Give Equivalent Exposures: Volatile F.P.s (curie sec/m³)	Gross F.P.s (curie sec/m³)
Lethal exposure	over 450R	over 350	over 400
Injury likely	100-450R	80-350	90-400
Injury unlikely, but some expense may be incurred. Observation required	25-100R	10-80	10-90
No injury or expense	less than 25R*	less than 10	less than 10

* 25R in one exposure or 50R in three months.

The first column indicates the equivalent whole body gamma radiation adopted as the basic criterion to define the several categories. Columns two and three have been calculated for these same criteria in terms of units used to estimate the effect of passage of the radioactive cloud. While these values are believed to be the best obtainable at the present time, many of the factors used in deriving them are highly uncertain. It should be noted that personal [fol. 4885] injury is considered to have occurred only in the first two categories. Expense might be incurred for exposures in the third category, but only for examination, observation and incidentals, not actual personal injury.

Degrees of Land Contamination

By far the largest dollar cost to the public of a major reactor accident would result from contamination of land areas by deposited fission products. Inhabitants of portions of the areas affected would have to be evacuated to avoid serious exposure. Access to various areas might be denied for different lengths of time, and the subsequent use of land for agricultural purposes might be curtailed, with possible loss of standing crops. The same basic exposure-injury criteria listed above (column 1) were used also for

determining the consequences of land contamination. Details of calculations are shown in Appendix D. In the case of land contamination, the existence of specific isotopes especially strontium[90], must be considered very carefully. The severe restrictions that might be imposed on farming arise almost entirely from the existence of this particular isotope.

To estimate the potential loss arising from problems of land contamination both the number of persons and the area affected were calculated. In some instances the costs were evaluated by associating them with an average cost per person. In the particular cases associated with farm restriction an average cost per square mile was used.
[fol. 4886] The categories chosen, and costs assumed for each are:

Range I.	Evacuation of personnel—immediate	$ 5000/person
Range II.	Evacuation of personnel—orderly and in a reasonable time	$ 5000/person
Range III.	Restrictions on land and outdoor activity	$ 750/person
Range IV.	Crop and farm restriction	$25,000/sq. mile

The criteria used in establishing these ranges are described in Appendix D. It should again be emphasized that they are based on meager data.

REACTOR ACCIDENTS ASSUMED

Three types of reactor accidents were considered necessary for this study in order to indicate the range of public hazard which could result and to delineate the influence of the important variables as described above on the magnitude of these hazards. The three "typical" cases selected are:

A. *The Contained Case*

In this accident, it is assumed that all of the fission products from the 500,000 Kw (thermal) reactor, after 180 days of operation, are released from the core

and distributed uniformly throughout the interior of the containment building. None is assumed to escape. The fission products are assumed to decay at their natural rate, with no attempt at decontamination, etc., after the accident. Hazard to the public would arise [fol. 4887] from the direct gamma radiation from the fission products dispersed inside the containment building. One inch of steel shielding by the walls of the building is assumed. The site boundaries are 2000 feet from the reactor.

B. *The Volatile Release Case*

In this case it is assumed that all of the volatile fission products in the reactor (500,000 Kw (thermal) after 180 days), i. e., xenon, krypton, iodine, bromine and 1% of the strontium are released from the containment building and are subsequently dispersed, with characteristics and meteorological conditions as described and specified above. See Appendix A.

C. *The 50% Release Case*

In this case, it is assumed that 50% of all fission products in the reactor (500,000 Kw (thermal) after 180 days) are released from the containment building and are subsequently dispersed, with characteristics and meteorological conditions as described and specified above. See Appendix A.

Each of these arbitrary cases represents a highly pessimistic assumption. Certainly more catastrophic releases of the Contained and the Volatile types are not possible. In the third type, it is conceivable that more than 50% of all fission products could be released, but this is considered to be so far in the realm of incredibility as not to merit consideration.

[fol. 4888] PART III

ESTIMATED CONSEQUENCES OF THE ASSUMED REACTOR ACCIDENTS

[fol. 4889]

ESTIMATED CONSEQUENCES OF THE ASSUMED REACTOR ACCIDENTS

In this part of the report, there is presented a brief summary of the calculated damages obtained from each of the assumed accidents, together with brief observations and pertinent comments on the results obtained in the respective cases. Reference is made to Appendices H and I, of Part IV, for more complete tabulation ~~of re~~sults.

Case I—The Contained Case

The assumption is made that all of the fission products are vaporized and dispersed within the containment shell. There is no release to the atmosphere. Damage to the public would then result from direct exposure to gamma radiation. The following tabular summary shows personal injuries and evacuation costs beyond the 2000 feet boundary of he reactor site.

	Personal Injury	
	Assuming Evacuation in 2 Hours (Persons)	Assuming Evacuation in 24 Hours (Persons)
Lethal exposure	0	0
Injury likely	0	6
Injury unlikely, but expense likely	1	15

Evacuation Costs		
Number of People	Area	Cost
67	1.8 sq. mi.	$335,000

[fol. 4890] *Observations and Remarks*

1. The above results would be the maximum possible for this type of accident in that all fission products would be involved and no shielding except the container is assumed.

2. Under the best conditions, namely, prompt evacuation of nearby personnel, no personal injury would be likely. The public loss would be due entirely to evacuation costs and payments for denial of use of land. This can be measured in the hundreds of thousands of dollars.

3. Under less favorable conditions, namely, slower evacuation, a small number of personal injuries might be expected.

4. Use of the typical site and population distribution is less satisfactory for this case since nearby population variations from site to site are larger than the numbers of people affected. The method does, however, give an order-of-magnitude.

5. For smaller site boundaries, larger numbers of people would be affected, especially in the injury category. However, with proper combinations of distance and shielding no loss to the public would be involved.

Case II—The Volatile Release

Here it was assumed that, because of a breach in the container or failure to close all openings, all volatile fission products would be discharged to the atmosphere at the time of the accident. Four different situations of meteorological conditions and two particle size distributions were considered. Furthermore, separate indication is given for releases which include 1 percent of the strontium inventory and for those which do not.

A full summary of the calculated damages is contained in Appendix I. The following table contains a brief summary to indicate the magnitude and range of the consequences.

[fol. 4891] *The Volatile Release Case*

Personal Injury

A. Lethal Exposure	Persons	Conditions at Release
Minimum	2	Temperature lapse
Maximum	900	Temperature inversion, 1 μ particles

Assuming that (1) the particle size distributions are equally probable, and (2) the distribution of weather conditions is as stated in Appendix I, then lethal exposures would be less than five people for those accidents which might occur during about one-half of the time or less than 300 people for those accidents which might occur during about three-fourths of the time.

B. Injury Likely	Persons	Conditions at Release
Minimum	10	Temperature lapse, 7 μ particles
Maximum	13,000	Temperature inversion, 1 μ particles

Using the same assumptions as under A, the number of persons injured would be less than 20 people for those accidents which might occur during about one-half of the time or 2000 people for those accidents which might occur during about three-fourths of the time.

Property Damage

II. Evacuation	Persons	Area (sq. mi.)	$ Millions	Conditions
Minimum	0	—	—	Temperature lapse, dry
Maximum	41,000	28	205	Temperature inversion, rain

[fol. 4892] Under the same assumptions as under A, the number of persons requiring evacuation would be less than 1000 people for accidents which might occur during about two-thirds of the time or 6000 people for those accidents which might occur during about nine-tenths of the time.

III. General Restrictions (due to Sr)	Persons	Area (sq. mi.)	$ Millions	Conditions
Minimum	20	1	.01	Temperature lapse, dry, 1 μ
Maximum	235,000	350	177	Temperature lapse, rain, 1 μ

Under the same assumptions as under A, the area placed under general restrictions would be less than 50 sq. mi. for those accidents which might occur during about three-fourths of the time.

IV. Agricultural Restrictions (due to Sr)	Area (sq. mi.)	$ Millions	Conditions
Minimum	3	.1	Temperature lapse, dry, 1 μ
Maximum	3,500	90.	Temperature lapse, rain, 1 μ

Under the same assumptions as under A, the area placed under agricultural restrictions would be less than 500 sq. mi. for those accidents which might occur during about nine-tenths of the time.

Observations and Remarks

1. The number of personal injuries is highly dependent upon existing weather conditions at the time of the accident. Few lethal exposures would occur during daytime conditions. Exposures of large numbers of persons would occur during temperature inversions, typical of nighttime conditions.

[fol. 4893] 2. Except when strontium accompanies the release, property damage would range from essentially no damage to cases of approximately two hundred million dollars. Without strontium, there would be no restrictions on agriculture.

3. The presence of strontium would add severe restrictions on land use both for general activity and for agricultural purposes. Decontamination would also be required within certain city areas. The net effect would be to increase the property damage and personal dislocation costs to a maximum of about 400 million dollars.

Case III—The 50 Percent Release Case

In this case it is assumed that 50 percent of all fission products would be released into the atmosphere and subsequently dispersed according to assumptions described earlier. Appendix I contains a summary of the personal injuries and property damages calculated for the variety of conditions considered. The table on the following page contains a brief summary to indicate the magnitude and range of the consequences.

[fol. 4894] *Personal Damage*

A. Lethal Exposure	Persons	Conditions at Release
Minimum	0	Hot release at any time
Maximum	3,400	Cold release, 1 μ particle size, temperature inversion

Assuming that (1) hot and cold releases are equally probable, (2) particle size distributions are also equally probable, and (3) the distribution of weather conditions is as stated in Appendix I, then lethal personal exposures would be less than 10 for accidents which might occur during about three-fourths of the time.

B. Injury Likely	Persons	Conditions at Release
Minimum	0	Hot release at any time
Maximum	43,000	Cold release, 1 μ particle size temperature inversion, dry

Using the same assumptions as under A, the number of persons injured would be less than 100 for accidents which might occur during about three-fourths of the time.

Property Damage

II. Evacuation	Persons	Area (sq. mi.)	$ Millions	Conditions
Minimum	0	0	0	Hot, temperature inversion
Maximum	460,000	760	2,300	Cold, 1 μ, temperature inversion, rain

Using the same assumptions as under A, the number of persons to be evacuated would be less than 50,000 for accidents which might occur during about three-fourths of the time.

III. General Restrictions	Persons	Area (sq. mi.)	$ Millions	Conditions
Minimum	0	0	0	Hot, 1 μ, dry
Maximum	3,800,000	8,200	[illegible],800	1 μ, rain

Using the same assumptions as under A, the area placed under general restrictions would be less than 1200 sq. miles for accidents which might occur during about three-fourths of the time.

IV. Agricultural Restrictions	Area (sq. mi.)	$ Millions	Conditions
Minimum	18	.5	Hot, 1μ, day, dry
Maximum	150,000	4,000.	Hot, 1μ, day, rain

Using the same assumptions as under A, the area placed under agricultural restrictions would be less than 10,000 sq. miles for accidents which might occur during about 93 percent of the time.

(The numbers above are from different cases and hence are not additive).

[fol. 4895] *Observations and Remarks*

The numbers shown in the previous summary are calculated on the basis of what we believe to be the best available assumptions, data and mathematical methods. As has been stressed elsewhere, there is considerable uncertainty about many of the factors, techniques and data, so that these numbers are only rough approximations. Where information is sufficiently complete we have chosen values to represent the most probable situation but where high degrees of uncertainty exist we have chosen values believed to be on the pessimistic (high hazard) side. The results shown would be quite sensitive to variations in some of the factors

which were used. As an example, the amount of fission products actually retained in people's lungs might be quite different from the amount assumed and this difference would change all the personal injury numbers greatly.

In addition there could be weather conditions which, when combined with other imaginable extremely adverse conditions, could result in damages greater than the maximum considered in this study.

The damages calculated for the assumed 50 percent fission product release would vary widely depending upon weather conditions and assumed temperatures of the released materials.

The lethal exposures could range from none to a calculated maximum of 3400. This maximum could only occur under the adverse combination of several conditions which would exist for not more than 10 percent of the time and probably much less.

[fol. 4896] Under the assumed accident conditions, the number of persons that could be injured could range from none to a maximum of 43,000. This high number of injuries could only occur under an adverse combination of conditions which would exist for not more than 10 percent of the time and probably much less.

Depending upon the weather conditions and temperature of the released fission products for the assumed accident, the property damage could be as low as about one-half million and as high as about seven billion dollars. For the assumed conditions under which there might be some moderate restrictions on the use of land or crops (Range IV), the areas affected could range from about 18 square miles to about 150,000 square miles.

[fol. 4897] PART IV

APPENDICES

[fol. 4899] APPENDIX A

The Nature and Extent of a Fission Product Release from a Power Reactor

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[fol. 4900] APPENDIX A

The Nature and Extent of a Fission Product Release from a Power Reactor

Introduction

The principal danger associated with the operation of nuclear reactors of any type is the possible release of the radioactive fission products which they contain. Power reactors are hazardous in this respect because, for economic reasons, they must be operated for long irradiation times and at high power levels. These circumstances lead to large accumulations of fission products. The danger associated with an explosive nuclear energy release in a reactor is quite mild in comparison to the potential hazards from these materials if they should be dispersed. Even in the worst imaginable cases of nuclear runaway the energy release would be comparable only to a mild chemical explosion. Chemical reactions occurring in the wake of a nuclear runaway might in fact contribute more energy than the runaway itself. If power reactors are located at sites similar to those now being proposed, the release of energy accompanying a reactor accident would constitute a negligible hazard to the public. The energy release is important only because it contributes to the possible extent of the fission product release.

[fol. 4901] *Basic Power Reactor Types*

Power reactors may be classified according to three basic constituents: moderator, fuel, and coolant. A few pertinent comments about the principal types may serve to suggest some of the complexities, differences and similarities

relevant to basic problems of safety. Fast reactors are designed in such a way that fissions are caused primarily by the absorption of fast neutrons by the fissionable material. Consequently, these reactors contain only weakly moderating materials. Thermal reactors contain strongly moderating substances such as graphite and water which greatly reduce the energy of the neutrons before they are absorbed. The fissionable fuel may be distributed throughout the reactor in the form of solid rods or plates, in which case the reactor is called heterogeneous, or it may be dispersed in the coolant fluid, in which case the reactor is called homogeneous. Most of the power reactors proposed to date are of the heterogeneous type. Water or liquid metals are among the chief materials now used to extract heat from a power reactor. When water is the moderator it serves also as the coolant. Examples are the pressurized water, the boiling water, and the aqueous homogeneous designs. Fast reactors and graphite-moderated thermal reactors generally utilize liquid metal coolants.

The reactor types differ markedly in engineering design, and each type poses its own peculiar safety problems. Some designs might be more prone to certain types of accidents than others, but it would be exceedingly difficult to compare [fol. 4902] the various power reactor types with regard to safety. It can be expected that before any reactor is approved for construction and operation all known problems relating to safety will have been resolved. In particular, such reactors would be expected to be inherently stable and would be operated according to certain prescribed procedures. They would be equipped with certain customary safety devices, as well as such special ones as their peculiarities dictate. All power reactors would contain substantially the same fission product inventory when operated under similar conditions. While the hazards posed by various reactor types may not be identical, they are at least similar in a number of respects.

Types of Reactor Accidents

Reactors can malfunction in many ways, and in this respect they are no different from other machines. Among other things, such malfunctioning could result from human

errors, equipment failures, design errors, and acts of God. Accidents resulting from such malfunctionings could result in power plant "outage" and damage to the reactor. Only a few types of accidents could plausibly lead to a release of fission products to the atmosphere. Two such accidents, a nuclear runaway and a loss of sufficient coolant to uncover the reactor core, are considered below to illustrate some of the complexities involved.

[fol. 4903] *The Nuclear Runaway*

A nuclear runaway would result if the reactor were made super-critical and all safety instrumentation failed to function. As a consequence, the reactor power and temperature would increase until the runaway were terminated either by the inherent self-stabilizing influence of the reactor or by actual mutilation of the reactor core. A possible consequence of an unchecked runaway could be the meltdown or vaporization of fuel elements and the release of fission products. Another possible consequence could be the initiation of exothermic chemical reactions between certain metals and liquids in the system. Such reactions would assist in the release and dispersal of fission products. However, it is highly improbable that the nuclear and the chemical energy release could cause much mechanical violence beyond the reactor shield. It is therefore feasible to build a gastight container around the reactor which would greatly reduce the chances of a fission product release to the atmosphere, if rupture of the reactor itself should occur.

The possibility of a serious nuclear runaway cannot be completely ruled out, but its occurrence can be made extremely unlikely by careful operating procedure, by adequate design, and by a multiplicity of control devices.

If a nuclear runaway were to occur, its effects would be minimized if the reactor had been designed to be inherently stable. The property of inherent stability implies that the [fol. 4904] production of heat causes physical changes within the reactor which reduce the reactivity. An inherently stable reactor will be self-regulating as soon as, or very shortly after, its temperature begins to rise. Water-

moderated reactors generally possess this self-regulating property to a marked degree, and it seems likely that the property can be designed into all types of reactors to at least some degree.

An inherently stable reactor is not completely immune to destructive runaways, however. In the 1954 Borax experiment it was possible substantially to wreck a stable boiling water reactor by deliberate introduction of a large amount of reactivity at a rapid rate. The self-stabilizing features in reactors may not always operate concurrently with the release of heat by the fission process but may be delayed. If a substantial reactivity were to be introduced into the reactor during this "delay", the reactor would behave essentially as though it were non-self-stabilizing and destruction could be the consequence.

Such large, rapid additions of reactivity are not easily achieved, and in a normally operating reactor could only occur if a series of unlikely misoperations or failures took place. No feature in the design of a reactor receives more attention than those which are incorporated to prevent such inadvertent reactivity additions. Design features and mechanical safeguards, in addition to the inherent self-stabilizing characteristics, are always incorporated to prevent such addition, and these must fail before a potentially hazardous situation would exist.

[fol. 4905] *The Loss of Coolant Accident*

A second major type of accident is the loss of coolant. Such an accident could result from a break in the primary coolant circulating system or from a rupture of the reactor vessel itself. Loss of coolant would permit the radioactive decay heat to melt the uncooled fuel, even though the nuclear reaction had stopped, and thereby permit release of volatile fission products. Calculation indicates that, at least for certain reactor core configurations, further heating of the fuel to the boiling point is precluded by radiation losses.[3] There is the additional possibility that the overheated fuel would react chemically with air entering the reactor, or, in the case of water-moderated reactors, with such water as remains. Such a reaction, if violent, would

help disperse the fission products and might furnish enough energy to break the external reactor container.

Even in the event of a major loop break it is possible to prevent a fuel meltdown by providing for emergency cooling of the core. This may be accomplished by maintaining a large tank of coolant. In any case, the principal line of defense against loss of coolant accidents would be adequate design and care in construction.

The consequences of a loss of coolant could be serious. But the event is highly improbable since it requires the occurrence of an unlikely material failure in the primary [fol. 4906] loop coupled with the unlikely failure of emergency cooling schemes, or else the unlikely failure of the reactor vessel itself.

Chemical Reactions

It has already been mentioned that excessive heating of the reactor through either nuclear runaway or loss of coolant could result in potentially violent chemical reactions. Three principal reactions are: sodium reacting with air, fuel metal reacting with air, and water reacting with fuel metal. The additional possibility exists that hydrogen evolved in the last reaction could react with oxygen.

The first reaction would occur if, as a result of an accident with a sodium cooled reactor, vaporized sodium came in contact with air. The reaction would take place as a rapid but non-violent burning of a vaporized sodium. The only effect of this burning would be to increase the pressure in the reactor's vapor container. Since the vapor container would be designed to withstand the pressure increase resulting from the burning of all the sodium in the reactor, this particular reaction would not be expected to cause a container rupture.

The second reaction would take place if air entered a ruptured reactor vessel and came into contact with hot fuel elements. The result would be rapid oxidation or burning of the metal. The reaction would be non-violent, but it could release a substantial portion of the fission products.

[fol. 4907] The third type of reaction, which is peculiar to heterogeneous water-moderated reactors, would be the only potentially violent one. The metals employed in the construction of fuel elements which would be reactive at high temperatures are zirconium and aluminum and possibly uranium. The total chemical energy available for these water-metal reactions equals or exceeds the energy that would be released in the worst possible nuclear excursion. However, the conditions for anything like a complete reaction would be difficult to achieve. Experience with water-metal reactions in reactors is at present almost totally lacking; therefore, conclusions must be based on information acquired from foundry practice and a few experiments.

The available information on the aluminum-water reaction may be briefly summarized as follows. In aluminum foundry practice, water is frequently used as a quench to form ingots from molten metal. This practice has infrequently led to violent explosions.[5] The occurrence of these explosions has been found to depend very sensitively on such conditions as the depth of the water, the diameter of the molten stream, and the presence of impurities. For example, a coating of grease on the water container was found to prevent the explosion, while iron rust was found to increase the tendency for explosion. Weils and West[6] at Argonne National Laboratory performed the experiment of pouring molten aluminum into water without obtaining an explosive reaction. Molten aluminum was also poured into water in an experiment at the Aerojet General [fol. 4908] Corporation Laboratories. The experiment was then modified by using a blasting cap to disperse the metal. Explosions failed to occur in either case; only the formation of an oxide film took place.[7] The conclusion to be drawn from the Argonne and the Aerojet General work is that a violent explosion will not occur under the special conditions of these experiments. Finally, it should be mentioned that, in the destructive Borax experiment, a meltdown of aluminum-clad fuel elements failed to produce an explosive water-metal reaction.[8]

Experiments performed at Westinghouse,[9] Aerojet General[7], and North American Aviation[10] indicate that the

zirconium-water reaction can be either a rapid oxidation or a violent explosion, depending on whether the zirconium is in massive form or finely dispersed. In the first case the reaction becomes noticeable at about 1200°C; well below the melting point of zirconium. In the presence of water the reaction becomes noticeable at about 1200°C, well below of heat is removed; while in the presence of steam the reaction is expected to proceed autocatalytically; i.e., the reaction, once started, will proceed without the application of external heat. In the experiments with dispersed zirconium and water the dispersal was brought about either by detonation of a blasting cap below the surface of the water while molten zirconium was poured in or by explosion of zirconium wires in water by means of rapidly discharging condensers. In either case the zirconium present reacted more or less completely with explosive violence.

[fol. 4909] A theoretical analysis has been made at Westinghouse[4] to determine the maximum possible extent of a water-metal reaction occurring in a pressurized water reactor. It was hypothesized that a major break had occurred in the coolant loop, resulting in loss of water and the complete uncovering of the reactor core. The temperature of the zirconium-uranium fuel elements would soon rise as a result of the fission product decay heat. When the metal reached 1200°C, the fuel elements would begin to react with the steam present in the core. The reaction would then proceed autocatalytically until the metal temperature was brought to the melting point. The melting would take place slowly releasing droplets which would fall into the remaining water below. At this point the water-metal reaction would be quickly quenched. By using experimentally determined heating curves for the water-zirconium system, a calculation was made of the amount of zirconium that could react from the inception of the reaction to the time of its quenching by heat losses from the metal droplets to the water. The maximum possible percentage of the metal which could react in this most favorable case was estimated to be 25 percent.

In the course of a water-metal reaction, hydrogen gas would be evolved which could react with oxygen after leaving the reactor. If the hydrogen exceeds a certain

critical concentration, an explosion is possible. But a very substantial amount of hydrogen would be required to raise the hydrogen concentration in the vapor container to this [fol. 4910] critical level. A certain quantity of hydrogen would be produced within the reactor as a result of its usual operation, but this amount is so small that it would constitute no explosive hazard.[1] Since the evolution of hydrogen from the water-metal reaction would take place slowly, it could be burned before explosive concentrations are reached. In some installations to insure that hydrogen burns as it is evolved, a number of electric igniters are located throughout the vapor container.

To summarize, the chemical reaction which poses the most serious danger in the event of a reactor accident is the water-metal reaction. This reaction, however, would be expected to proceed as a vigorous but incomplete oxidation of the metal at elevated temperatures. In the case of zirconium, it is expected that no more than 25 percent of the metal would react. A violent and more or less complete reaction of the metal would require the metal to be finely dispersed. Such a dispersal could occur only as a result of fuel vaporization, which was previously pointed out to be a highly unlikely event.

The Function of Vapor Containers

Since the energy release (whether chemical, nuclear, or both) which might accompany a reactor accident is expected to be of comparatively mild intensity, it is feasible to construct a steel shell to confine the fission products which might escape from the reactor. Because of the large volume [fol. 4911] of such a shell, it can be readily designed to withstand the pressure loading resulting from accidents capable of rupturing the reactor vessel. Presumably it would be impractical to design such a vapor container to confine the worst conceivable accidents. It is designed rather to contain all credible accidents. For example, the vapor container for a pressurized water reactor would be designed to withstand either the pressure resulting from a water release or 25 percent of the energy available for a chemical reaction, but not both simultaneously. In this case,

calculation[4] indicates that the pressure produced by the first event would be relieved through heat losses from the container before the second event could take place.

There is always the possibility that the vapor container could be penetrated by flying fragments resulting from failures in the system. The use of ductile metals in construction would greatly reduce the probability of such failures and therefore the probability of missile formation. In addition, the resistance of the vapor container to the penetration by missiles could be increased by lining the inside of the shell with a layer of reinforced concrete.

While the vapor shell could probably not withstand severe shock-wave effects, it is considered extremely unlikely that such shock phenomena could be initiated by either a nuclear or a chemical energy release. The speed of a nuclear excursion would be limited by the lack of means of introducing reactivity rapidly into the reactor system, while the speed of a chemical energy release would probably be governed [fol. 4912] by the rate of mixing of the reactants. In either case the energy release could be expected to be much slower and less destructive than an equivalent energy release from a detonating explosive. Energy releases calculated for reactor accidents are sometimes expressed in TNT weight equivalents. Such comparisons ignore the fact that the rates of energy release in the two cases may be greatly different. The damage in the reactor thus is overestimated.

Thus the vapor container surrounding a reactor may be considered another line of defense for the protection of the public. These structures are not impregnable, but they are designed to be capable of confining the accidents which can be regarded as credible.

The Extent of Fission Product Release

The question is now raised: in the highly unlikely event of a reactor accident which leads to a rupture in both the reactor vessel and its vapor container, what would be the expected percentage release of the fission products? The answer to this question depends in a complicated way on the details of the accident. Various possible accident situations could lead to different amounts of fission product release, e.g., fuel meltdown unaccompanied by a chemical reaction; meltdown followed by a non-violent oxidation of

the metal by water; meltdown in the presence of air accompanied by combustion of the metal; and either violent chemical reaction or vaporization, or both.

[fol. 4913] The first situation would require that no water be present in the reactor and that no combustion take place. The latter requirement could be met if the fuel elements have a melting temperature well below the temperature required for rapid combustion. In experiments performed at Oak Ridge National Laboratory, Parker[11] has electrically melted uranium-aluminum fuel elements in the presence of air without causing combustion. It can be reasonably assumed that in an accident situation molten fuel metal will quickly assume a physical shape which is no longer conducive to the molten state; e.g., it can form into drops which fall and resolidify. In the previously mentioned experiments Parker observed that, if the irradiated molten uranium-aluminum fuel element is refrozen within a few seconds, about 60 percent of the noble gases, xenon and krypton, leave the metal in addition to about 25 percent of the iodine. The percentage of the bromine escaping can be reasonably assumed to equal that of the iodine. Once resolidification of the fuel has taken place, the escape of radioactivity would be expected to stop. While these fission products make up the bulk of the released radioactivity, minute quantities of less volatile substances such as tellurium were also detected to have escaped from the metal. It is reasonable to assume that other metals having similar volatility, such as strontium, could escape in minute quantities as well.

[fol. 4914] In the second situation, the release of the more volatile fission products is assisted by the oxidation of the metal by water. Two hydrogen atoms are released for every atom of metal oxidized, and, in the course of escaping, the evolved hydrogen disrupts from the lattice atoms of the more volatile elements, which likewise escape. Experiments have been carried out at the Westinghouse Atomic Power Division to determine what products are released as a result of the corrosion by water of irradiated uranium metal.[12] These experiments were performed at relatively low temperatures (600°F). The fairly volatile metals cesium and rubidium were observed to escape quantitatively, while the less volatile barium and the biologically important element

strontium were found to escape only to the extent of 5 percent. It seems reasonable to assume that the gaseous elements, halogens and the noble gases, would also escape quantitatively although no determination was made. Unfortunately, such data are not available for the corrosion of more typical reactor fuels such as uranium-zirconium alloy at more realistic temperatures. In any case, the evolution and escape of hydrogen gas are expected to govern the release of fission products. It can be argued that the behavior of the hydrogen should not depend strongly on the metal being oxidized and that therefore the fission product release observed for uranium is a likely one for other reactor fuels as well. According to Westinghouse estimates, a maximum of 25 percent of the zirconium-[fol. 4915] uranium fuel could be oxidized by water. Therefore, the maximum expected release of strontium in case of a complete fuel meltdown in the presence of water would be 5 percent of 25 percent or 1 percent, on the assumption that negligible amounts of strontium are released from the unreacted metal. On the basis of Parker's data the same release could be expected to include 70 percent of the noble gases and 44 percent of the halogens as well as less important percentages of some of the volatile metals.

The third situation could occur if the fuel element melting temperature were high enough for combustion to accompany melting. The process of combustion would involve considerable disruption of the oxidizing material and would cause the release of a substantial fraction of the fission products. Parker observed that uranium-stainless steel fuel elements burned vigorously after rapid heating to 2000°C and that 50 percent of the total gamma activity of the fuel element was removed as a result. If all the noble gases and halogens are assumed to escape, it could be inferred that 25 percent of the remaining fission products were removed. Zirconium-uranium alloy has a high melting temperature (about 1800°C), therefore, it might be a candidate for combustion. Parker, however, observed that zirconium-uranium fuel elements heated slowly to the melting point did not burn. Such slow heating should reasonably simulate the melting of fuel elements by decay heat.

[fol. 4916] Actually, in the case of a pressurized water reactor it is highly doubtful that air could even enter the

core while the fuel was in the molten state, the reason being that steam at greater than atmospheric pressure would fill the core for a matter of hours following the loop failure. It is conceivable in some reactor designs that air could enter the core following a coolant loop rupture; however, in view of Parker's observations it appears doubtful that combustion would occur.

The final situation could lead to substantial dispersal of the fuel. Whether this would significantly augment the release of fission products is open to question. It is reasonable to assume that fission products would escape quantitatively from the metal that had reacted or vaporized; but it is doubtful whether major portions of fuel could vaporize or react violently even in the unlikely event that these two processes did take place. The dispersed metal should behave substantially as in the first and second situations already discussed. Thus, even in the case of violent disruption within the reactor, the fission product release should not be expected to exceed substantially the release in the case of fuel combustion.

Conclusions

On the basis of the best available information, the mechanism fission product release most likely to occur, appear to be either a fuel meltdown or a meltdown accompanied by an oxidation of fuel by water, depending on whether or not water is present in the reactor. In the former case, the [fol. 4917] release would be confined to about half the noble gases and about a quarter of the halogens contained in the fuel. In the latter case, the release would consist primarily of a somewhat more complete release of these same volatiles in addition to approximately 1 percent of the contained strontium.

A meltdown followed by combustion could result in a release of 50 percent of the contained radioactivity. A conservative guess would be that a like percentage of strontium would be released. In the light of experimental evidence this type of release seems less likely than either of the first two. Finally, a violent release could not reasonably be expected to exceed 50 percent. Such a mode of release would also be unlikely.

Speculation has so far been concerned only with the escape of fission products from reactor fuel and has not taken into account the condensation and absorption of these substances on metal surfaces in the reactor and vapor container during their passage to the atmosphere. The preceding estimates are therefore somewhat conservative, at least in the case of the less volatile fission products; but since these estimates involve uncertainties, there is some justification for conservatism.

[fol. 4918] *References*

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[fol. 4920] APPENDIX B

Description of Reactor and Site

[fol. 4921] APPENDIX B

Description of Reactor and Site

Description of Reactor

The reactor chosen for this study is a 500,000 thermal kilowatt reactor. This power rating is in the range of the ratings of the large power reactors now proposed. However, since all the cost analyses that have been performed on reactors show that the cost per kilowatt-hour of electricity decreases with increase in reactor power, it is expected that the power level of future reactors will tend to be large.

The reactor is assumed to be fueled with Uranium 235. Also, the fuel reprocessing cycle is taken to be 180 days. This time interval seems appropriate for reactors now proposed.

Description of Site

It is assumed in this report that the reactor would be 30 miles from a large city, and located near a large body of water. It is logical to place the reactor near the users of power, since transmission costs are proportional to distance; on the other hand, land costs are less outside the city than inside. Nearness to a water supply is postulated because water is necessary for steam condensation. All the power reactor sites proposed to date are within 30 to 40 miles of a city and near an adequate water source.

[fol. 4922] Choice of a typical distribution of population around a reactor site was arrived at by a consideration of the actual distribution around five reactor sites. It develops that the total population within given radial distances (R) of less than 30 miles is remarkably similar for these sites. This population can be calculated by the expression Population $= 200\ R^{2.83}$ where R is in miles. Figure 1 shows the

population curves for three government controlled sites and two proposed private sites with the above population equation plotted. At medium distances the actual populations around the proposed commercial sites are lower than those calculated from this equation by a factor of up to 4. This is about the same as azimuthal variation. It should be noted that the government sites have been in existence for some time. It would be expected that more people and factories would move into the region 10 to 20 miles from a new reactor as time goes by, so that the areas around commercial sites would become like those around the government sites, for which the population equation is quite good.

For distances greater than 30 miles, population density is assumed to be 500 people per square mile rather than the United States average of 55 per square mile. The states with high population densities have been deliberately chosen because power reactors are expected to be built in regions where there are many power users. Examples of population densities in some industrial states as given in "The Statistical Abstract of the United States, 1955", are:

[fol. 4923]

Population Density, people/mi²

	1940	1950
Rhode Island	674	748
New Jersey	553	643
Massachusetts	545	596
New York	281	309

The above numbers show that population densities are increasing at a very rapid rate.

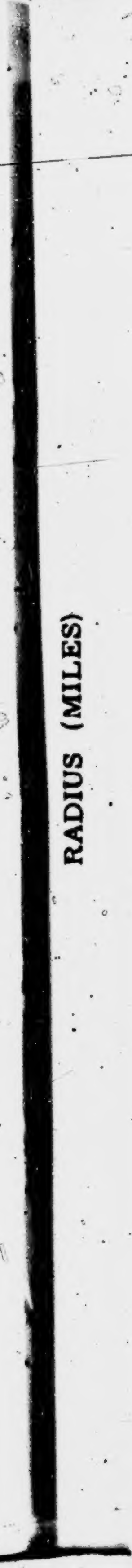
The choice of the characteristics of the nearby city was difficult. For purposes of calculation ~~the city was~~ assumed to have one million people uniformly dispersed over a circular area of 15-kilometer (about 10 miles) radius. While no city of this exact description exists, it is felt that such characteristics provide a reasonable basis for the calculation of hypothetical damages.

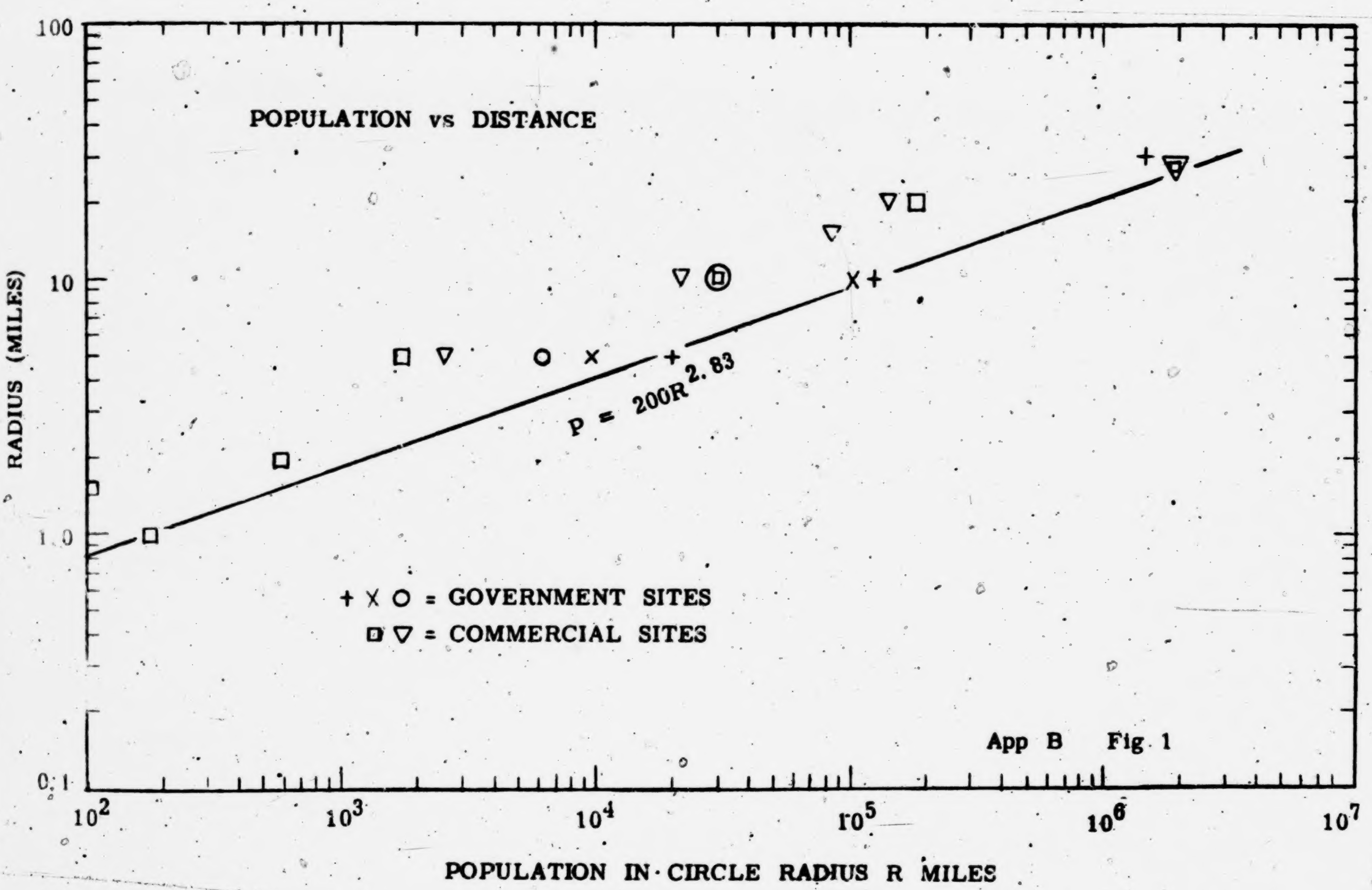
Figures

Figure 1. Population vs. Distance.

[fol. 4924]

(See opposite)☞





POPULATION vs DISTANCE
RADIUS (MILES)
100
10
1.0
0.1
10^2
10^3
10^4
10^5
10^6
10^7
$P = 200R^{2.83}$
+ X O = GOVERNMENT SITES
□ ▽ = COMMERCIAL SITES
App B Fig 1
POPULATION IN CIRCLE RADIUS R MILES

4524

[fol. 6846]

Before the Atomic Energy Commission

Docket No. F-16

In the Matter of

Power Reactor Development Company

Opinion and Initial Decision

This matter comes before the Commission on an application by the Power Reactor Development Company (hereinafter called PRDC) for a facilities license under Section 104(b)[1] of the Atomic Energy Act of 1954, as amended, 42, U.S.C., Sec. 2134(b). The application was filed January 6, 1956 and, as amended by ten amendments, requests a class 104(b) license for a period of twenty-five years for a fast breeder reactor system described in the application. The precise question before the Commission is whether the provisional construction permit of August 4, 1956, arising from the license application and described below, should be continued, modified, suspended, revoked or otherwise disposed of. The reactor as described would have a rated capacity of 300,000 thermal kilowatts and 100,000 electric kilowatts. The proposed location of the plant is at Lagoona Beach, approximately thirty miles south of Detroit, Michigan, on land leased from the Detroit Edison Company. Notice of the filing of the application was published in the Federal Register on June 27, 1956.

[fol. 6847] PRDC is a non-profit membership corporation organized under the laws of the State of Michigan. Its membership includes public utilities and equipment manufacturers interested in the advance and development of the peaceful use of atomic energy. By contract it has engaged the technical services of Atomic Power Development Associates, Inc. (hereinafter called APDA), which is a non-profit membership company organized under the laws

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[1] Quoted *infra*, page 13.

of the State of New York. APDA's organizational structure includes companies and individuals engaged in the study of fast breeder technology. The memberships of PRDC and APDA are largely duplicatory; thirteen of the twenty-one members of PRDC, including all but two of the utility members, are members of APDA and a fourteenth PRDC member represents four member operating companies of APDA. The president of Detroit Edison Company is also president of PRDC and APDA.

The contract between PRDC and APDA provides for APDA to furnish PRDC with the complete design for the proposed reactor and with technical services in connection with reactor components. The agreement likewise provides for APDA to procure and assemble the "component test facility" and to conduct extensive non-nuclear tests with it. Upon completion of those tests APDA will transfer to PRDC that facility, which for practical purposes is the proposed reactor with dummy fuel and one-third of the primary coolant loops.

Detroit Edison Company has agreed with PRDC to buy the steam generated by the reactor and to furnish PRDC with housekeeping services. Detroit Edison plans to install conventional steam generating facilities in association with the reactor and to introduce the electric power so generated into its regular power net.

[fol. 6848] The license application states that the earliest date for completion, in the sense of finishing construction to the point of introducing nuclear fuel, of the reactor is December 15, 1959, and the latest date is December 15, 1960. Statements in the record make it clear that construction is now expected to be finished about September of 1960. According to the license application, the member companies of PRDC had committed themselves, subject to call, to provide financing which was expected to be sufficient, when augmented by some $15 million of loans which had been arranged with banks, and some further commitments made by member companies, to cover the anticipated cost of construction and operation of the proposed plant. The application requested an allocation of nuclear material for fuel sufficient to cover an operating period of twenty-five years.

On August 4, 1956 the Commission issued a conditional or provisional construction permit for the proposed project, and reserved nuclear materials for it, but it took no action with respect to a license to operate the plant. Notice of the issuance of the construction permit was given by publication in the *Federal Register* (21 FR 5974). The permit specifically provided that conversion to a license depended upon satisfactorily showing that the final design of the reactor gave reasonable assurance that the health and safety of the public would not be adversely affected by its operation. The permit further provided for its expiration unless within one year the Applicant provided sufficient data to enable the Commission to make the findings of financial qualification required by the Atomic Energy Act and the Commission's regulations. The Commission, on August 1, 1957, extended the time for this submission and for the evaluation of that evidence.

[fol. 6849] On August 31, 1956 the International Union, United Automobile, Aircraft, and Agricultural Implement Workers of America, affiliated with the American Federation of Labor-Congress of Industrial Organizations, and Walter P. Reuther, Emil Mazey, and Carlos Gastambide, described as officers and members of the UAW, petitioned to intervene, sought a formal hearing, and requested the Commission to suspend the construction permit theretofore issued. Similar petitions were filed by the International Union, United Paper Workers of America, affiliated with the American Federation of Labor-Congress of Industrial Organizations, and six individuals identified as officers and members thereof, and by the International Union of Electrical, Radio and Machine Workers, likewise affiliated with the AFL-CIO, and three individuals described as officers and members thereof (these three labor unions and their officers and members referred to will hereafter be collectively called "Interveners"). The petitions to intervene were granted on October 8, 1956. On the same date the Commission issued a notice of hearing, order and memorandum directing that a hearing be held before Jay A. Kyle, Examiner, on issues designated by the Commission. Those issues were set forth by the Commission as follows:

A. 1. Whether there is information sufficient to provide reasonable assurance that a utilization facility of the general type proposed in the application can be constructed and operated at the location proposed therein without undue risk to the health and safety of the public.

[fol. 6850] 2. Whether there is reasonable assurance that technical information omitted from and required to complete the application will be supplied.

B. Whether, pursuant to Section 50.40(b) of the Commission's regulations, the Applicant is financially qualified to engage in the proposed activities; and whether, pursuant to Section 50.60(c)(2) of the Commission's regulations, the Applicant is financially qualified to receive an allocation of special nuclear material.

C. With respect to any matter in controversy in this proceeding, whether the Commission may, or should, grant any exemption pursuant to Section 50.12 of its regulations on the ground that such exemption is "authorized by law and will not endanger life or property or the common defense and security and . . . [is] otherwise in the public interest."

D. If the issues in the proceeding are resolved in favor of continuing the construction permit, what additional or different provisions, if any, should be incorporated in it.

The Commission's order provided that the Examiner should certify the record to the Commission at the conclusion of the hearing, and the Examiner did so on November 22, 1957. Answers were filed to the notice of hearing by both PRDC and the Interveners, and various motions were likewise filed and appropriately disposed of; they were considered in orders issued by the Commission on October 3 and 8 and December 12, 1956. There was a [fol. 6851] prehearing conference on November 29, 1956, and the hearing lasted, with intermittent recesses, from January 8, 1957 to August 7, 1957. In the meanwhile, on

March 4, 1957, the State of Michigan, through its Attorney General, petitioned to intervene and participate in the proceedings. Intervention was granted to the State of Michigan, but intervention was denied to one Elliott Earl who sought intervention on August 15, 1957. The State of Michigan has never taken any firm position with respect to the construction permit but has only emphasized its responsibility for the health, safety and welfare of its citizens.

Much of the testimony in the hearing was submitted in written narrative form in accordance with the permission granted by an order of the hearing Examiner issued November 29, 1956. Objection by the Interveners to the submission of written narrative testimony was overruled by the Examiner on January 29, 1957, and the Commission on February 28, 1957 denied an interlocutory appeal from this order.

In addition to the testimony elicited in the course of the hearing, statements were received from the AFL-CIO, State of Michigan, and the Cooperative League of the U.S.A., and these statements were included in the record. Briefs were subsequently filed by the Applicant and the Interveners, and the cause was argued orally before the Commission on May 29, 1958. The Commission has considered the complete record of the evidence and the briefs and arguments presented.

In addition to contentions regarding lack of due process, the Interveners, in their petition for intervention, alleged that the Commission violated the Atomic Energy Act in granting a conditional construction permit, and based that conclusion on four grounds which remain the foundation of their position through brief and oral argument. These four grounds, as stated in the Interveners brief, were:

> [fol. 6852] (a) The Commission violated Section 185 of the Act by failing to make a finding requisite to the grant of a construction permit on a provisional basis, under Section 50.35, Title 10 CFR, that the Commission is satisfied that it has information sufficient to provide reasonable assurance that a utilization facility of the general type proposed can be constructed and operated

at the proposed location without undue risk to the health and safety of the public; and

(b) The Commission violated Section 185 of the Act, as implemented and interpreted by Section 50.40(b), Title 10 CFR, by failing to make a finding that PRDC is financially qualified to engage in the proposed activities in accordance with Commission regulations; and

(c) The Commission violated Section 185 of the Act, as implemented and interpreted by Section 50.60(b), Title 10 CFR, by failing to make a finding that PRDC is financially qualified to assume responsibilities for payment of Commission charges for materials, and to undertake and carry out the proposed use of special nuclear materials for a reasonable period of time; and

(d) The Commission violated Section 185 of the Act and the aforesaid regulations by making an allocation of special nuclear material to PRDC without the requisite finding of financial qualifications.

[fol. 6853] These grounds were summarized on oral argument by counsel for the Interveners as the inherent lack of safety of the reactor as designed, its special hazard in the selected location, and the financial inability of the Applicants to construct and operate the reactor.

[fol. 6854] A brief description of the proposed reactor will show that it is the largest of this particular type so far proposed to be erected in the United States. The reactor vessel and core would be housed in a reactor building, which would in turn be housed in a containment structure, 120 feet high and 72 feet in diameter, of welded carbon steel. The reactor vessel is welded stainless steel, 36 feet high, with a diameter, enclosing the core and blanket, of 9.5 feet in the lower section and of 14.5 feet in the upper section. For biological shielding, as well as other purposes, this vessel would contain twelve inches of stainless steel in the lower portion, thirty inches of borated graphite between the lower reactor vessel and the primary shield tank, and a 3.5 foot concrete wall enclosing the reactor vessel and the primary coolant system; and a seven foot concrete wall

and five feet of concrete and steel flooring would enclose the entire reactor compartment. The core of the reactor is to be a cylinder approximately 30.5 inches in diameter by 31.2 inches high, consisting of uranium pins 27% enriched in the isotope U-235, clad in zirconium and held in square stainless steel subassemblies. A blanket of uranium depleted in the isotope U-235 would surround this core and would comprise, with the core, a cylinder 78.5 inches in diameter and 67 inches long. The reactor is designed to be cooled by 5000 cubic feet of liquid sodium, pumped by electric pumps upward through the core and blanket, thence through three primary coolant loops, to heat-exchangers, and then back to the reactor vessel, with a secondary coolant system of liquid sodium in three loops between the primary coolant and the steam generator where its heat is converted into steam. This dual system is provided to prevent the chemical reactions that would take place if the radioactive sodium [fol. 6855] in the primary system had a chance to reach the water in the steam generator.

Classification of reactors may be made in several ways, but one ground for classification is the speed of the utilized neutrons. In the chain reaction process, neutrons are released by the fission of radioactive atomic nuclei and strike the nuclei of other fissionable atoms, which in turn fission and release their own neutrons against other atoms in the continuing process. When a Uranium-235 nucleus fissions, it releases its neutrons at an average speed of about 10,000 miles per second. If the design of the reactor is such that it can utilize the neutrons moving at or near this speed, it is called a "fast" reactor. If, however, the reactor employs as a "moderator" one of the materials, such as for example, graphite, either ordinary or heavy water, or beryllium, which has the ability to slow down neutrons to a speed of about one mile per second without capturing those neutrons, and if it employs those relatively slow moving nuclear particles in the chain reaction, it is called a "thermal" reactor.

Not all the neutrons released by the fissioning atom produce more fissions; some neutrons escape, and others are captured by fertile but non-fissionable atoms like Uranium-238, the isotope which constitutes 99.3% of all the uranium

in nature. Instead of stabilizing as Uranium-239, however, such an atom passes through a brief existence in the form of Neptunium-239 and then stabilizes as plutonium which [fol. 6856] is itself fissionable. If a reactor is so designed that its Uranium-238 atoms capture neutrons and therefore change into plutonium atoms at a rate greater than the rate at which the Uranium-235 atoms are fissioning, it is called a "breeder." The proposed PRDC reactor is so designed, and its conversion ratio is 1.20, or, in other words, it is designed to produce 20% more fissionable material than it consumes.

The safety and control mechanism for the reactor core includes rods containing Boron 10, which reduces radioactivity by absorbing neutrons. These rods control the radioactivity in the core by their insertion into or withdrawal from the core, and they are designed so that in the event of a mechanical or electrical failure, their magnetic grips will release them to drop into the core.

There would also be associated buildings for such purposes as storage of new and spent fuel elements, w[illegible] storage, clean-up and storage of sodium and inert gas, repair shop, laboratory, general services, holding basins for liquid wastes, etc.

As in the case of any other power reactor at this stage of the art, in addition to the design features of the reactor which will be determined and demonstrated before the actual construction of the reactor, many will be determined and demonstrated during its actual construction.

[fol. 6857] Congress, in the Atomic Energy Act, provided for two types of licenses: the Section 103[2] class of license

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[2] Sec. 103. Commercial Licenses.

a. Subsequent to a finding by the Commission as required in section 102, the Commission may issue licenses to transfer or receive in interstate commerce, manufacture, produce, transfer, acquire, possess, use, import, or export under the terms of an agreement for cooperation arranged pursuant to section 123, such type of utilization or production facility. Such licenses shall be issued in accordance with the provisions of chapter 16 and subject to such conditions as the Commission may by rule or regulation establish to effectuate the purposes and provisions of this Act.

b. The Commission shall issue such licenses on a non-exclusive basis to persons applying therefor (1) whose proposed activities

requires a finding that the proposed utilization or production facility has been sufficiently developed to be of practical [fol. 6858] value for industrial or commercial purposes; the Section 104[3] class of license, in addition to the medical

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will serve a useful purpose proportionate to the quantities of special nuclear material or source material to be utilized; (2) who are equipped to observe and who agree to observe such safety standards to protect health and to minimize danger to life or property as the Commission may by rule establish; and (3) who agree to make available to the Commission such technical information and data concerning activities under such licenses as the Commission may determine necessary to promote the common defense and security and to protect the health and safety of the public. All such information may be used by the Commission only for the purposes of the common defense and security and to protect the health and safety of the public.

c. Each such license shall be issued for a specified period, as determined by the Commission, depending on the type of activity to be licensed, but not exceeding forty years, and may be renewed upon the expiration of such period.

d. No license under this section may be given to any person for activities which are not under or within the jurisdiction of the United States, except for the export of production or utilization facilities under terms of an agreement for cooperation arranged pursuant to section 123, or except under the provisions of section 109. No license may be issued to an alien or any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. In any event, no license may be issued to any person within the United States if, in the opinion of the Commission, the issuance of a license to such person would be inimical to the common defense and security or to the health and safety of the public.

[3] Sec. 104. Medical Therapy and Research and Development.

a. The Commission is authorized to issue licenses to persons applying therefor for utilization facilities for use in medical therapy. In issuing such licenses the Commission is directed to permit the widest amount of effective medical therapy possible with the amount of special nuclear material available for such purposes and to impose the minimum amount of regulation consistent with its obligations under this Act to promote the common defense and security and to protect the health and safety of the public.

b. The Commission is authorized to issue licenses to persons applying therefor for utilization and production facilities involved in the conduct of research and development activities leading to the demonstration of the practical value of such facilities for in-

[fol. 6859] therapy provisions not pertinent to this case, is of a research and development character and envisions progress toward practical applications. The rules and regulations of the Commission with regard to licenses do not make a distinction between Section 103 and Section 104 licenses because Section 185[4] of the Act, providing for

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dustrial or commercial purposes. In issuing licenses under this subsection, the Commission shall impose the minimum amount of such regulations and terms of license as will permit the Commission to fulfill its obligations under this Act to promote the common defense and security and to protect the health and safety of the public and will be compatible with the regulations and terms of license which would apply in the event that a commercial license were later to be issued pursuant to section 103 for that type of facility. In issuing such licenses, priority shall be given to those activities which will, in the opinion of the Commission, lead to major advance in the application of atomic energy for industrial or commercial purposes.

c. The Commission is authorized to issue licenses to persons applying therefor for utilization and production facilities useful in the conduct of research and development activities of the types specified in section 31 and which are not facilities of the type specified in subsection 104b. The Commission is directed to impose only such minimum amount of regulation of the licensee as the Commission finds will permit the Commission to fulfill its obligations under this Act to promote the common defense and security and to protect the health and safety of the public and will permit the conduct of widespread and diverse research and development.

d. No license under this section may be given to any person for activities which are not under or within the jurisdiction of the United States, except for the export of production or utilization facilities under terms of an agreement for cooperation arranged pursuant to section 123 or except under the provisions of section 109. No license may be issued to any corporation or other entity if the Commission knows or has reason to believe it is owned, controlled, or dominated by an alien, a foreign corporation, or a foreign government. In any event, no license may be issued to any person within the United States if, in the opinion of the Commission, the issuance of a license to such person would be inimical to the common defense and security or to the health and safety of the public.

[4] Sec. 185. Construction Permits.

All applicants for licenses to construct or modify production or utilization facilities shall, if the application is otherwise acceptable to the Commission, be initially granted a construction permit. The construction permit shall state the earliest and latest dates for the

construction permits, was intended by the Congress to apply to both Section 103 and Section 104 licenses. It thus appears that our rules are designed to provide for Commission determination of the form and scope of a construction permit as appropriate in a particular case, depending upon the particular state of research and development currently applicable to the proposed project.

[fol. 6860] The Commission has published regulations pursuant to its authority under the statute, and certain portions of those regulations have a bearing on the safety issue in this case. The basic statement of the standards under which the Commission will issue licenses is Regulation 50.40 (10 CFR 50.40).[5]

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completion of the construction or modification. Unless the construction or modification of the facility is completed by the completion date, the construction permit shall expire, and all rights thereunder be forfeited, unless upon good cause shown, the Commission extends the completion date. Upon the completion of the construction or modification of the facility, upon the filing of any additional information needed to bring the original application up to date, and upon finding that the facility authorized has been constructed and will operate in conformity with the application as amended and in conformity with the provisions of this Act and of the rules and regulations of the Commission, and in the absence of any good cause being shown to the Commission why the granting of a license would not be in accordance with the provisions of this Act, the Commission shall thereupon issue a license to the applicant. For all other purposes of this Act, a construction permit is deemed to be a "license".

[5] 50.40 *Common Standards*. In determining that a license will be issued to an applicant, the Commission will be guided by the following considerations:

(a) The processes to be performed, the operating procedures, the facility and equipment, the use of the facility, and other technical specifications, or the proposals in regard to any of the foregoing collectively provide reasonable assurance that the applicant will comply with the regulations in this chapter, including the regulations in Part 20, and that the health and safety of the public will not be endangered.

(b) The applicant is technically and financially qualified to engage in the proposed activities in accordance with the regulations in this chapter.

(c) The issuance of a license to the applicant will not, in the opinion of the Commission, be inimical to the common defense and security or to the health and safety of the public.

Also relevant on the standards for issuing licenses, and showing that the construction permit is a step toward a license rather than the equivalent of it, is Regulation 50.45 (10 CFR 50.45).

Actual issuance of both licenses and construction permits is covered by Regulation 50.50 (10 CFR 50.50).

[fol. 6861] But the regulation most vitally involved in the present issue is Regulation 50.35 (10 CFR 50.35).

The position of the Interveners in their brief and on oral argument was that issuance of the construction permit violated the Commission's own regulations and that therefore its issuance was illegal. This position was founded on an interpretation of the statute and regulations to the effect that all the same findings as the Commission would have to

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50.45 *Standards for construction permits.* An applicant for a license or an amendment of a license who proposes to construct or alter a production or utilization facility will be initially granted a construction permit, if the application is in conformity with and acceptable under the criteria of subsection 50.31 through 50.38 and the standards of subsection 50.40 through 50.43.

50.50 *Issuance of licenses and construction permits.* Upon determination that an application for a license meets the standards and requirements of the act and regulations, and that notification, if any, to other agencies or bodies have been duly made, the Commission will issue a license, or if appropriate a construction permit, in such form and containing such conditions and limitations including technical specifications, as it deems appropriate and necessary.

50.35 *Extended time for providing technical information.* Where, because of the nature of a proposed project, an applicant is not in a position to supply initially all of the technical information otherwise required to complete the application, he shall indicate the reason, the items or kinds of information omitted, and the approximate times when such data will be produced. If the Commission is satisfied that it has information sufficient to provide reasonable assurance that a facility of the general type proposed can be constructed and operated at the proposed location without undue risk to the health and safety of the public and that the omitted information will be supplied, it may process the application and issue a construction permit on a provisional basis without the omitted information subject to its later production and an evaluation by the Commission that the final design provides reasonable assurance that the health and safety of the public will not be endangered.

make in order to issue a license must also be made in order to issue a construction permit. The Interveners in other words maintain that insofar as its factual foundation is concerned, the distinction between a construction permit and a license is only in the name, and that unless there is sufficient foundation in the record now for the issuance of a license to operate, the construction permit should be suspended and construction should stop. With this interpretation we cannot agree.

[fol. 6862] There can be no doubt that public safety is the first, last, and a permanent consideration in any decision on the issuance of a construction permit or a license to operate a nuclear facility. The Commission regards the importance of public safety so highly that it considers that it does not lose jurisdiction of this subject even after a license has been issued, at any stage in the course of its construction, or, for that matter, even after a facility is in operation. Although not very likely, newly discovered scientific knowledge could conceivably indicate an unanticipated hazard even after some years of operation of a facility, and we regard the Commission in such circumstances to be obliged by the law to take appropriate action.[9]

Sections 104 and 185 of the statute, and pertinent regulations, particularly Section 50.35, on their face, however, negate the synonymity of the construction permit and the license to operate, and their meaning is clear. The very fact that the construction permit is referred to as the natural precursor of the license emphasizes the distinction between them, and the distinction is well established in the systems of other administrative agencies.

[fol. 6863] Everyone agrees, including the Interveners in their brief and on oral argument, that there is no hazard to the public in the construction of the reactor beyond that which is inherent in any other heavy construction. The only

[9] Although our knowledge of it is limited to press reports, we recall the recent experience of a British manufactured turbine powered aircraft type, which, although fully licensed on the basis of the usual exhaustive tests by the appropriate British authority, after transporting a great many commercial passengers over a period of months, proved to be structurally deficient and therefore hazardous. All aircraft of this particular type were promptly grounded, and they never returned to commercial service.

question of hazard to the public clearly rises when the time arrives for the reactor to go critical. The issue as seen by the Interveners, however, is that construction of the reactor inevitably means its operation. This contention is based on the argument that the heavy investment in the reactor will generate irresistible pressure for its operation so as to protect the investment itself; corollary is the position that it would be unfair to PRDC to permit it to make a multimillion dollar investment and then deny it a license to operate. As the Interveners see the problem, issuance of the license is automatic—virtually a ministerial act—once the construction permit has been issued.

The Commission certainly does not so interpret the Act or its regulations. Perhaps it may be argued that in the case of types of reactors which have become virtually production line-items, such as certain relatively small research reactors of standard types, nothing short of a wholly unanticipated event of major significance would cause the Commission to withhold an operating license to a purchaser after issuance of a construction permit. In the case of the PRDC reactor design, however, PRDC has been on notice since before the first shovel of dirt was moved that its construction permit is provisional upon further demonstration of many technological facts, including the complete safety of the reactor. The Applicant has, in fact, stated on the record that it is going to proceed at first under only the [fol. 6864] construction permit so as to develop a background that will support reasonable assurance that the reactor can be operated in complete safety, and has stated its intention to seek operating authority only after all necessary facts have been assembled. Since PRDC has recognized the experimental nature of the reactor it is building and since it has expressly waived any commitment (if there exists any of the type that the Interveners contend is implied by the construction permit), for an operating license, the possibility that the Commission would be in any way bound cannot be visualized. It would be hard to imagine a case where an applicant would be less able to argue that he had been misled by previous favorable Commission action. Under the circumstances of this case, moreover, and in view of the wording of the provisional construction permit,

it must be perfectly clear that PRDC is assuming a substantial financial risk with its eyes wide open, and that the generation of any pressure from such ingredients would be quite absurd. An operating license may be considered in light of the clear wording of the statute and on the basis of the legislative history, as a logical, but not automatic, sequel to the fulfillment of the conditions of the construction permit, but we emphasize in this decision the importance of the present construction permit provisions requiring the demonstration of safety and of satisfying those conditions.

Several problems having a relationship to the operational safety of the reactor must be kept in mind. First there is the possible instability of the reactor, second is the possibility of a fuel element melt-down and its reassembly into a critical mass of dangerous proportions; third is the possible damage resulting from the leak of radioactive fission products in the event of a rupture of the steel contain-[fol. 6865] ment shell of the reactor as a result of one of the previously described accidents of some other.

First of all the Commission must agree that the possibility of one of these accidents—or some other—cannot be categorically excluded. If the statute and the regulations are to be interpreted, as the Interveners imply, so that the Commission must be *certain* that such an accident will never occur, then no licensed reactor could ever be built. The very fact that Section 104, under which developmental reactors can be licensed, is in the statute reveals that such a standard was never contemplated, and the concept of reasonable assurance of safety must be sensibly, though severly applied.

The spectrum of expert testimony on the safety of the proposed PRDC design was surprisingly narrow. The report of the Reactor Safeguards Committee referred to in Interveners' brief can be literally accepted by the Commission, as it was by many of the witnesses, as meaning that the state of human knowledge at the time the report was prepared would not support an absolute guarantee that there would be no safety problem in the operation of the reactor; but it also permits the conclusion that going forward with the construction phase of this project would, by the very nature of the information developed in the course

of evolving design, help to remove doubt concerning safety and would tend to provide an increasingly firm foundation for the reasonable assurance required by the statute that the project could be *operated* without undue risk to public health and safety. Of one thing there can be certainty: until the questions raised by the Reactor Safeguards Committee have been answered, to the satisfaction of the Commission, there will be no license to operate the PRDC reactor.

[fol. 6866] It is in the nature of reactor design, although certainly not unique to it, that many features remain to be designed and demonstrated after construction is begun—and, indeed, some features redesigned and replaced after operation is under way—but the Commission foresees the solution of such problems as still remain. Certainly there is nothing in the design as presently contemplated that is known to be inherently and immediately dangerous, and no insoluble problems are presently identified.

It is enough for the purposes of the present proceeding, that is for provisional construction permit purposes, and for satisfaction of the requirements of the statute and the regulations, that there be reasonable assurance that the reactor can eventually be operated safely. We conclude that the present state of knowledge as described in the record give that assurance and that the accident possibilities considered on the record do not negate that assurance; on the contrary, we anticipate that the growth of present knowledge within a time reasonable in relation to the production schedule will continue to strengthen it.

The question of safety, however, obviously cannot be considered without regard to proposed location. Just as the concept of reasonable speed on a flat and straight stretch of a transcontinental highway is quite different from reasonable speed on a mountain road or a city street, so also the operation of a reactor at Lagoona Beach, Michigan cannot be equated to operation of one at the National Reactor Testing Station in Idaho.

We are satisfied on the basis of the record that the likelihood of any breach of the containment shell which PRDC proposes, designed as it is to contain an explosive force equivalent to 600 pounds of TNT, a limit beyond any known

[fol. 6867] accident possibility in the reactor, is extremely remote, but the possible consequences in the event of such a rupture should still not be ignored. If the containment structure failed to hold the fission products of the reactor as a result of a nuclear accident, the consequences would depend on the meteorological conditions at the moment as well as upon the nature of the fission products released. This problem is not unique to either the PRDC reactor or to fast breeders in general; rather it applies to all types of reactors.

With regard to location, however, the record shows that the Applicant is well aware of its problem and is attempting to take all reasonable steps not only to prevent accidents but also to curtail the consequences of an accident if there should be one. Studies of weather, hydrology, geology, etc., have yielded considerable information, and are still in progress. That the data of these types are not complete or conclusive may readily be agreed, but such data as are available in the record tend to support the finding that there is nothing peculiar to the proposed location that negates a conclusion that reasonable assurance of the safe operation of the reactor will be as likely in that location as elsewhere. The record gives reasonable assurance to that effect, and we anticipate that knowledge to be acquired will fortify it.

In the case of the proposed site of the reactor, as in the case of the inherent safety of the proposed reactor itself, we emphasize that the degree of certitude that satisfies us for purposes of the provisional construction permit would not be the same as we would require if we were at this [fol. 6868] moment considering the issuance of the operating license. "Reasonable assurance" can be a different standard for the one purpose from what it is for the other. Even the Interveners on oral argument seemed to acknowledge the differing degrees of certitude that the two situations would require, and we think that the form of our order adequately protects all participants in this proceeding, as well as the public as a whole, in its manner of providing for the distribution of new information that is expected to be useful when the time arrives for consideration of an operating license.

The financial qualification of an Applicant is by the Commission's regulations made an explicit condition of the issuance of the license. The requirement is stated in Regulation 50.40 (b) quoted above, and it is repeated in connection with the treatment of allocation of special nuclear materials in Regulation 50.60 (b):

"The request for incorporation of [provisions designating quantities of special nuclear material available for use by the facility] may be made simultaneously with the submission of an application for construction permit of facility license or at any time thereafter. Such request should be accompanied by at least the following information:

(1) The applicant's financial qualifications to assume responsibility for payment of Commission charges for the materials, and to undertake and carry out the proposed use of special nuclear material for a reasonable period of time;

* * *

[fol. 6869] (c) A request for the incorporation in a construction permit or license of provisions designating the amount of special material available for use by the facility will be approved by the Commission if:

* * *

(2) The applicant appears to be financially qualified to assume responsibility for the payment of Commission charges for the material and to undertake and carry out the proposed use of special nuclear material for a reasonable period of time;

* * *"

There is a substantial amount of evidence in the record concerning the financial history and prospects of PRDC and its members. We have already noted that public utilities and equipment manufacturers of financial position, having among them, in fact, a gross equity of about two billion dollars, are included among those members, and they have contributed upon call some $23 million to PRDC.

So far as the record shows, funds have been forthcoming from the members of PRDC on call, and there is no evidence to indicate this willingness to continue to support PRDC will cease. PRDC appears to be paying its bills, and its credit has not been indicated to be questionable.

It appears in the record that the utilities which have contributed to PRDC have so far had their payments allowed by the commissions regulating them in their respective States as proper expenditures. In addition, the availability to PRDC of a $15 million loan with a New York bank appears firm. All in all the record shows that PRDC has assets available to it of nearly $49,500,000, and this sum seems to be substantial evidence upon which to base a conclusion [fol. 6870] of financial responsibility. Present estimates of the cost of the reactor are in the neighborhood of $34 million, and one may estimate that research and development expenses, administrative expense, interest during the construction period, start-up, and other miscellaneous items will add about $8 million to that figure.

The Interveners contended that a good many of the items included in the total cost of the reactor had been underestimated, and that therefore the total was grossly underestimated. We agree that the experience with reactors has been such that cost estimates have customarily been exceeded, and we are prepared to agree that many items will cost more than applicant has estimated—but probably not as much more as Interveners contend. Likewise we are prepared to agree that the applicant has over-estimated the revenues it is likely to receive from the sale of plutonium. It appears to us, however, that the financial history of PRDC and the financial standing of its members are such that the increases in cost feared by the Interveners will be covered through the devices that PRDC has previously utilized.

The Commission's regulations obviously do not require an applicant to have cash on hand to cover all possible contingencies of costs higher and revenues lower than estimates. Cash on hand or in the bank is certainly prime evidence of financial responsibility but it is clearly not the only evidence. On the basis of the financial estimates in the record, and even after making allowances for reasonable

cost over-runs, and revenue under-runs, such as will probably be the product of plutonium prices lower than estimated by PRDC, we are satisfied that the prima facie showing and the reasonable probability of financial qualification [fol. 6871] to construct and operate the proposed reactor, to receive an allocation of nuclear materials and to pay Commission charges and prices therefor for at least a reasonable time after scheduled start-up of the reactor has not been rebutted. The Intervener's argument and the Applicant's not too strenuous disagreement that the project is likely to be a financial failure for the PRDC membership does not refute it within the framework of the issues of this proceeding.

We find no need to discuss or to pass upon the question of exemption from the Commission's regulations.

The argument of the Interveners to the effect that the ruling of the Examiner permitting the introduction of direct testimony in the form of written narrative was error is without merit. While there may be substantial division of opinion among members of the Bar concerning the practice of presenting direct testimony by written narrative rather than orally, nevertheless, the practice is so well established in administrative proceedings as to admit no doubt of its validity.

The contention of the Interveners that they were denied a fair hearing because certain consultants of the Commission who intended to appear as witnesses were advised of the existence of conflict of interest laws is without merit. The letter to the consultants was in no way characterized by duress; rather it merely alerted those consultants in a very tactful way to the possible desirability of consulting their own counsel. The consultants were free to act, and the choice of whether or not to do so was completely their

[fol. 6872] The objections of the Interveners to the participation of former Chairman Lewis L. Strauss in the decision and their motion to disqualify him became moot when Mr. Strauss' term as a Commissioner expired on June 30, 1958.

The Interveners' contention that they were denied a fair hearing because they did not have access to Restricted Data and refused to follow the normal clearance procedures is

likewise without merit. The Interveners showed no proof of prejudice or injury, and the legal duty of the Commission to classify certain material as Restricted Data and to establish a system for access to it is readily apparent from the Atomic Energy Act.

The Commission has considered all the other points raised by the Interveners in their briefs and an oral argument and finds them without merit.

The remaining part of the Opinion and Initial Decision of the Commission of December 10, 1958, including numbered findings concurring opinion of Commissioner Graham, order and construction permit as amended, are printed at pages 607-29 of Volume II of the printed record.

[fol. 6909]

BEFORE THE ATOMIC ENERGY COMMISSION

Docket No. F-16

In the Matter of

POWER REACTOR DEVELOPMENT COMPANY

INTERVENORS' EXCEPTIONS TO OPINION AND INITIAL DECISION, WITH SUPPORTING BRIEF

Benjamin C. Sigal, 1126 Sixteenth Street, N.W., Washington 6, D.C., Attorney for Intervenors.

Of Counsel: Harold A. Cranefield, Lowell Goerlich.

January 14, 1959.

[fol. 6910]

UNITED STATES OF AMERICA

ATOMIC ENERGY COMMISSION

Docket No. F-16

In the Matter of

POWER REACTOR DEVELOPMENT COMPANY

INTERVENORS' EXCEPTIONS TO THE OPINION AND INITIAL DECISION OF THE ATOMIC ENERGY COMMISSION

The Commission erred in making the following findings and conclusions:

1. That the rules of the Commission are designed to provide for Commission determination of the form and scope of a construction permit as appropriate in a particular case, depending upon the particular state of research and development currently applicable to the proposed project (Op., p. 14).[1]

2. That the Intervenors interpreted the statute and regulations to the effect that all the same findings as the Commission would have to make in order to issue a license must also be made in order to issue a construction permit, and the distinction between the construction permit and the license is only in the name (Op., p. 16).

3. That since PRDC has recognized the experimental nature of the reactor it is building and since it has expressly waived any commitment for an operating license which may be implied from the issuance of the construction [fol. 6911] permit, the possibility that the Commission would be in any way bound cannot be visualized . . . An operating license may be considered in the light of the clear wording of the statute and the basis of the legislative history, as a logical, but not automatic sequel to the ful-

[1] References in this form are to page numbers of the Opinion and Initial Decision.

fillment of the conditions of the construction permit . . . (Op., p. 19).

4. That the Intervenors imply that the Commission must be *certain* that an accident will never occur before a license may be issued (Op., p. 20).

5. That the report of the Reactor Safeguards Committee means that the state of human knowledge at the time the report was prepared would not support an *absolute* guarantee that there would be no safety problem in the operation of the reactor, but it also permits the conclusion that going forward with the construction phase of this project would, by the very nature of the information developed in the course of evolving design, help to remove doubt concerning safety and would tend to provide an increasingly firm foundation for the reasonable assurance required by the statute that the project would be *operated* without undue risk to the public health and safety (Op., p. 20).

6. That it is enough for the purposes of the provisional construction permit, and for the satisfaction of the requirements of the statute and regulations, that there be reasonable assurance that the reactor can eventually be operated safely (Op., p. 21).

7. That in the case of the proposed site of the reactor, the degree of certitude that satisfies the Commission for purposes of the provisional construction permit would not be the same as would be required if the Commission were now considering the issuance of the operating license (Op., p. 22).

[fol. 6912] 8. That the Intervenors on oral argument seemed to acknowledge the differing degrees of certitude which the two situations would require (Op., p. 23).

9. Funds have been forthcoming from the members of PRDC on call and there is no evidence to indicate this willingness to continue to support PRDC will cease (Op., p. 24) (T. 346, 508, 509).[2]

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[2] References in this form are to pages of the record supporting the exception.

10. That while the Commission agrees that the experience with reactors has been such that cost estimates have customarily been exceeded, and is prepared to agree that many items will cost more than PRDC estimated, and is prepared to agree, further, that PRDC has overestimated the revenues it is likely to receive from the sale of plutonium, nevertheless the financial history of PRDC and the financial standing of its members are such that increases in cost feared by the Intervenors will be covered by the devices that PRDC has previously utilized (Op., p. 25) (T. 3434, 3435, 3438, 792, 794).

11. That the objection of the Intervenors to the ruling of the Examiner permitting the introduction of direct testimony in the form of written narrative is without merit (Op., p. 26).

12. That the contention of the Intervenors that they were denied a fair hearing because certain consultants of the Commission who intended to appear as witnesses were advised of the conflict of interest laws is without merit (Op., p. 26).

13. The Intervenors' contention that they were denied a fair hearing because they did not have access to restricted [fol. 6913] data because they refused to follow the normal clearance procedures is without merit (Op., p. 27).

14. That the operations of PRDC will, if conducted according to the construction permit issued by the Commission on August 4, 1956, be subject to the Atomic Energy Act of 1954, as amended, and the Rules and Regulations issued by the Commission (No. 1).

15. That there is no inherent hazard and danger to the health and safety of the public in the operation of fast breeder reactors (No. 5) (T. 2937, 3129, 3130).

16. That reactors using sodium as coolant have operated successfully and have demonstrated the ability to achieve satisfactory purity of sodium and to avoid significant corrosion (No. 9).

17. That the proposed experiments with EBR I, Mark III core are expected, if properly carried out, definitively

to establish the correctness or incorrectness of the bowing hypotheses as the cause of the positive temperature coefficient observed in EBR I (No. 14).

18. That although many questions remain to be answered regarding the problem of meltdown and disassembly or reassembly of the core or a substantial portion thereof, the existence of these problems does not negate the likelihood that they will be answered in due course; and in time for the answers to precede any decision of the Commission with regard to an operating license for the proposed PRDC reactor (No. 15) (T. 2040, 2952, 2953, 3131, 2927, 3102, 3103, 3132).

19. That the necessity for constructing a prototype of the proposed reactor at a remote location has not been shown, but if the program of meltdown investigation should prove inconclusive it will be necessary to reconsider the question of need for a prototype (No. 17).

[fol. 6914] 20. That there is reasonable assurance that theoretical and experimental investigations which have been undertaken, together with operating experience at one or more of the EBR I, EBR II and Dounreay reactors, will establish definitively, prior to the scheduled completion date of the PRDC reactor, whether or not the reactor proposed by PRDC can be so operated; it is probable that evidence will establish that the reactor proposed by PRDC can be so operated (No. 18) (T. 2999, 3109, 3111, 3130).

21. That design modifications required by the results of studies of the meteorology, lake currents, air diffusion, flooding, and other problems can be considered at the time that the question of a license to operate is before the Commission (No. 20) (T. 3340, 3341).

22. That the suitability of the site for the proposed reactor depends upon the inherent safety of the reactor and the demonstration that no credible accident can release significant quantities of fission products into the atmosphere. If these are established, and there is reasonable assurance that they can be, it is probable that the site will prove suitable for the proposed reactor, but a definitive

evaluation of the suitability of the proposed site cannot be made at the present time (No. 21).

23. That the Commission finds reasonable assarance in the record that a utilization facility of the general type proposed in the PRDC application and the amendments thereto can be constructed and will be able to be operated at the location proposed without undue risk to the health and safety of the public (No. 22).

[fol. 6915] 24. That the member companies of PRDC should provide additional funds, if necessary, to take care of adverse contingencies during construction and test operation of the proposed reactor (No. 23) (T. 346, 508, 509).

25. That the record does not demonstrate that a cost over-run on construction and research and development will exceed PRDC's available assets by any amount and especially by an amount large enough to disprove PRDC's financial qualifications (No. 24) (T. 3407, 3408, 3473, 336-338; 3287-3326).

26. That PRDC expects to cover any loss of special nuclear material resulting from an accident by insurance, application of unencumbered assets or a guarantee from its sponsors, and has, therefore, shown a reasonable probability of being able to pay Commission charges for loss of special nuclear material through 1963 (No. 25) (T. 792-794).

27. That while data presently available indicate that a reactor of the general type described in the Application can be so designed that no credible accident in the course of its operation is likely to result in the release of significant quantities of fission products into the atmosphere, this conclusion has not been demonstrated sufficiently at this time to justify issuance of an operating license (No. 30) (T. 2933, 3118; Benedict Narr. T. 8, 10, 12).

28. That theoretical and experimental programs now under way should establish the foregoing conclusion sufficiently to justify issuance of an operating license. Results of these programs should be available prior to the time it should be necessary for the Commission to rule on the oper-

ating aspect of the license application (No. 31) (T. 3130, 2999, 3269, 3270, 3083).

[fol. 6916] 29. That the proposed site is generally suitable for a reactor of the type and size described in the Application, if the reactor is otherwise shown to be capable of operation without undue hazard, including demonstrations of stability and adequate containment (No. 32) (Wolman, Narr. 13, 15, 16, 17, 20; T. 3340, 3341).

30. That there is reasonable assurance that technical information omitted from the Application and required to complete it will be supplied prior to the time when it is necessary for the Commission to rule on the license application itself (No. 33) (T. 2040, 2927, 2952, 3102-3, 3109, 3114, 3132).

31. That PRDC is financially qualified to engage in construction and operation of the reactor described in the Application and to receive the allocation of special nuclear material therefor (No. 34).

32. That the issuance of construction permit will not be inimical to the common defense and security or to the health and safety of the public (No. 35).

33. The Commission erred in failing to make any findings with respect to the legality of the issuance of the conditional construction permit on August 4, 1956. The Commission erred, specifically, in failing to find:

a. That on August 4, 1956, the Commission did not have sufficient information to provide reasonable assurance that a facility of the general type proposed could be constructed and operated at Lagoona Beach, Michigan without undue risk to the health and safety of the public;

b. That on August 4, 1956, the Commission did not have sufficient information to provide reasonable assurance that the technical information omitted from and required to complete PRDC's application for a license would be supplied;

[fol. 6917] c. That on August 4, 1956, PRDC was not financially qualified to engage in the proposed activities in accordance with the Commission's regulations.

d. That on August 4, 1956, PRDC was not financially qualified to assume responsibility for payment of Commission charges for material and to undertake and carry out the proposed use for the material for a reasonable time.

The Commission erred, further, in finding:

34. That there is no merit in the contention of the Intervenors that the Commission denied them a fair hearing by limiting the issues in the proceedings so as to preclude proof that issuance of a conditional construction permit to PRDC on August 4, 1956 was illegal (p. 27).

35. That there is no merit in the contention of the Intervenors that they were denied a fair hearing as a result of the Hearing Examiner's refusal to require AEC to state its position on the issues specified by the Commission (p. 27).

The Order, dated December 10, 1958, is erroneous (p. 42).

The page references to the record following various exceptions above indicate testimony which contradicts the findings referred to in said exceptions. Exceptions which are not followed by references to the record are based on provisions of the Act and Commission regulations adopted pursuant thereto, particularly Sections 185 and 189 (a) of the Act, and Sections 2.740, 50.30 to 50.38, 50.40, 50.45, 50.50, 50.55, and 50.60 (b) and (c) of the Commission's regulations; Section 7(a) of the Administrative Procedure Act; and the First and Fifth Amendments to the Constitution of the United States.

Respectfully submitted,

Benjamin C. Sigal, Attorney for Intervenors.

Of Counsel: Harold A. Cranefield, Lowell Goerlich.

[fol. 6918] [File endorsement omitted]

IN UNITED STATES COURT OF APPEALS
FOR THE DISTRICT OF COLUMBIA CIRCUIT

No. 15,271

INTERNATIONAL UNION OF ELECTRICAL, RADIO AND MACHINE WORKERS, AFL-CIO; UNITED AUTOMOBILE, AIRCRAFT AND AGRICULTURAL IMPLEMENT WORKERS OF AMERICA; and UNITED PAPERMAKERS AND PAPERWORKERS, Petitioners,

v.

UNITED STATES OF AMERICA, Respondent.

AMENDED PETITION FOR REVIEW OF AN ORDER OF THE ATOMIC ENERGY COMMISSION—Filed July 27, 1959

To the Honorable, the Judges of the United States Court of Appeals for the District of Columbia Circuit:

International Union of Electrical, Radio and Machine Workers, AFL-CIO; United Automobile, Aircraft and Agricultural Implement Workers of America; and United Papermakers and Paperworkers, Petitioners, respectfully petition this Court, pursuant to Atomic Energy Act of 1954, 68 Stat. 919, Section 189, Act of December 29, 1950, as amended, ch. 1189, 64 Stat. 1129, and Administrative Procedure Act, Section 10, as amended, 60 Stat. 243, to review an order of Atomic Energy Commission, hereinafter called Commission, affirming and continuing in effect, with certain modifications, a construction permit issued to Power Reactor Development Company for construction of an atomic power reactor. The proceeding resulting in said order is designated upon the records of the Commission as Matter of Power Reactor Development Company, Docket No. F-16.

1. A conditional construction permit was issued to Power Reactor Development Company by the Commission on

August 4, 1956, and the Petitioners intervened in this proceeding on August 31, 1956, alleging that the Commission had, by issuing said conditional construction permit, violated the Atomic Energy Act of 1954, 68 Stat. 919, *et seq.* and the Commission's regulations adopted pursuant thereto. Petitioners requested the Commission to suspend the permit pending a final determination of the proceedings, and on final order to rescind and declare a nullity said permit until such time as there had been full compliance by Power Reactor Development Company with all pertinent provisions of the Act and the Commission's regulations.

[fol. 6919] 2. After due proceedings before the Commission, the Commission issued its Opinion and Final Decision on May 26, 1959, affirming and continuing said construction permit, with certain modifications. A copy of said Opinion and Final Decision is attached hereto.

3. The findings and order of the Commission are not supported by substantial evidence on the record considered as a whole, and are contrary to applicable law, including Commission regulations.

4. This Court has jurisdiction of this petition, which is brought by parties aggrieved by the final order of the Commission, under the provisions of the Act of December 29, 1950, ch. 1189, Sections 3 and 4, 64 Stat. 1130.

Wherefore, the Petitioners pray that the Court take jurisdiction of this proceeding and of the questions determined therein, review the final order of the Commission, and make and enter, upon the basis of the record, a decree setting aside the order of the Commission.

Respectfully submitted,

Benjamin C. Sigal, Attorney for Petitioners.

July 25, 1959.

[fol. 6920] [File endorsement omitted]

IN UNITED STATES COURT OF APPEALS

FOR THE DISTRICT OF COLUMBIA CIRCUIT

[Title omitted]

ORDER ALLOWING INTERVENTION OF POWER REACTOR DEVELOPMENT COMPANY—August 24, 1959

Upon consideration of the motion of Power Reactor Development Company for leave to intervene in this case, and it appearing that no objections have been filed, it is

Ordered by the court that Power Reactor Development Company is allowed to intervene in this case.

Per Curiam.

Dated: August 24, 1959.

[fol. 6921] [File endorsement omitted]

IN UNITED STATES COURT OF APPEALS

FOR THE DISTRICT OF COLUMBIA CIRCUIT

[Title omitted]

ORDER ALLOWING INTERVENTION OF STATE OF MICHIGAN—October 30, 1959

Upon consideration of the motion of the State of Michigan for leave to intervene as a party respondent in this case, and it appearing that no objections have been filed, it is

Ordered by the court that the State of Michigan is allowed to intervene in this case.

Per Curiam.

Dated: October 30, 1959.

[fol. 6922] [File endorsement omitted]

IN UNITED STATES COURT OF APPEALS
FOR THE DISTRICT OF COLUMBIA CIRCUIT

[Title omitted]

STATEMENT OF ISSUES OF INTERVENOR POWER REACTOR DEVELOPMENT COMPANY—Filed November 14, 1959

Pursuant to the Orders of the Court herein dated October 22 and November 6, 1959, Intervenor Power Reactor Development Company herewith files the following statement of the issues which it desires to present:

1. Whether the order of the Atomic Energy Commission of May 26, 1959, continuing in effect the provisional construction permit of the Intervenor Power Reactor Development Company and providing procedure for continuing review of this project by the Commission and for a further hearing and determination with respect to both safety of operation of the proposed reactor and the financial qualification of the applicant, prior to issuance of any operating license, is a final order which aggrieves the Petitioners or otherwise so affects their rights as to make such order now reviewable at their instance, under the provisions of Section 189 of the Atomic Energy Act of 1954 as amended (42 U.S.C. §2239), the Judicial Review Act of December 29, 1950, as amended (5 U.S.C. §1031-42), and Section 10 of the Administrative Procedure Act (5 U.S.C. §1009).

[fol. 6923] 2. Whether the first, fourth and sixth numbered issues included in "Petitioners Statement of the Issues", filed with this Court November 3, 1959, are moot or otherwise not properly presented by the decision of the Atomic Energy Commission sought to be reviewed herein.

3. Whether the second, third and eighth numbered issues included in "Petitioners Statement of the Issues", filed with this Court November 3, 1959, are properly presented by the decision of the Atomic Energy Commission sought to be reviewed herein.

In accordance with Rule 17(c) of the Rules of this Court, Intervenor Power Reactor Development Company further reserves the right to present in its brief a counter-statement of the questions presented in a form other than that in which such questions are stated and argued in the brief to be submitted by Petitioners.

Respectfully submitted,

W. Graham Claytor, Jr., Attorney for Intervenor, Power Reactor Development Company.

Covington & Burling, Washington, D. C.; Miller, Canfield, Paddock and Stone, Detroit, Michigan, Of Counsel.

Dated: November 13, 1959.

[fol. 6924] Certificate of Service (omitted in printing).

[fol. 6925] [File endorsement omitted]

In United States Court of Appeals

For the District of Columbia Circuit

[Title omitted]

Statement of Issues for the Respondents—Filed November 13, 1959

Pursuant to the order of this Court dated October 22, 1959, respondents have examined the statements of issues tendered by the petitioners and the intervenor Power Reactor Development Company. Respondents have determined that they do not desire to present any further issues in addition to those already stated by the petitioners and intervenor.

Respondents reserve the right to set forth in their brief in this case a "Statement of Questions Presented", pur-

suant to Rule 17(c)(1) of the Rules of this Court, in a form other than stated by petitioners and intervenor.

Respectfully submitted,

Samuel D. Slade, Lionel Kestenbaum, Department of Justice, Washington 25, D.C., Attorneys for Respondents.

Loren K. Olson, General Counsel, Atomic Energy Commission, Washington 25, D.C.

[fol. 6926] CERTIFICATE OF SERVICE (omitted in printing).

[fol. 6927] [File endorsement omitted]

IN UNITED STATES COURT OF APPEALS

FOR THE DISTRICT OF COLUMBIA CIRCUIT

[Title omitted]

PETITIONERS' SUPPLEMENTARY STATEMENT OF THE ISSUES— Filed December 8, 1959

Pursuant to the Order of the Court herein, dated October 22, 1959, Petitioners herewith file a supplementary statement narrowing the issues, as follows:

1. Did the Commission violate the Atomic Energy Act of 1954, and its own regulations, by issuing a provisional construction permit to the Power Reactor Development Company, on August 4, 1956, without finding:

a) That there was information sufficient to provide reasonable assurance that a utilization facility of the general type proposed by the Company could be constructed and operated at the proposed site without undue risk to the health and safety of the public;

b) That there was reasonable assurance that technical information omitted from and required to complete the application would be supplied by the Company;

c) That the Company was financially qualified to engage in the proposed activities pursuant to Section 50.40 (b), Title 10 CFR; and

[fol. 6928] d) That the Company was financially qualified to receive an allocation of special nuclear material pursuant to Section 50.60 (c)(2), Title 10 CFR?

2. Did the Commission violate the Act, and its own regulations, by continuing in effect the construction permit issued to Power Reactor Development Company on the basis of a finding that there is reasonable assurance in the record, for the purposes of this provisional construction permit, that a utilization facility of the general type proposed by the Company can be constructed and operated without undue risk to the health and safety of the public?

3. Did the Commission violate the Act, and its regulations, by continuing in effect the construction permit issued to Power Reactor Development Company without a finding that, as of the date of said Decision and Order, there was information sufficient to provide reasonable assurance that a utilization facility of the general type proposed by the Company can be constructed and operated at the proposed site without undue risk to the health and safety of the public?

4. Did the Commission violate the Act, and its own regulations, by continuing in effect the construction permit issued to Power Reactor Development Company on the basis, *inter alia*, that for the purposes of a provisional construction permit, there is reasonable assurance that the Company is financially qualified to engage in the construction and operation of the reactor described in the PRDC application and to receive the allocation of special nuclear material therefor?

5. Is the finding of the Commission that, for the purposes of a provisional construction permit, there is reasonable assurance that the Power Reactor Development Company is financially qualified to engage in the construction and operation of the reactor described in its application, and to receive the allocation of nuclear material there-

for, supported by substantial evidence on the record considered as a whole?

[fol. 6929] 6. Did the Commission violate the Act, and its own regulations, by continuing in effect the construction permit issued to Power Reactor Development Company without finding that said Intervenor is financially qualified to engage in construction and operation of the reactor described in said Intervenor's application, and to receive an allocation of special nuclear material therefor?

Respectfully submitted,

Benjamin C. Sigal, 1126 Sixteenth Street, N. W., Washington 6, D. C., Attorney for Petitioners.

December 8, 1959

[fol. 6930] Certificate of Service (omitted in printing).

[fol. 6931] [File endorsement omitted]

In United States Court of Appeals

For the District of Columbia Circuit

No. 15271

International Union of Electrical, Radio and Machine Workers, AFL-CIO; United Automobile, Aircraft and Agricultural Implement Workers of America; and United Papermakers and Paperworkers, Petitioners,

—v.—

United States of America and Atomic Energy Commission, Respondents.

Power Reactor Development Company, State of Michigan, Intervenors.

On Petition to Review an Order of the Atomic Energy Commission

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OPINION—June 10, 1960

Mr. Benjamin C. Sigal for petitioners.

Mr. Lionel Kestenbaum, Attorney, Atomic Energy Commission, with whom Assistant Attorney General Doub, Messrs. Loren K. Olson, General Counsel, Atomic Energy Commission, Courts Oulahan, Special Assistant to the General Counsel, Atomic Energy Commission, and Samuel D. Slade, Attorney, Department of Justice, were on the brief, for respondents.

[fol. 6932] Mr. W. Graham Claytor, Jr., with whom Mr. John Lord O'Brian was on the brief, for intervenor Power Reactor Development Company.

Mr. Jerome Maslowski entered an appearance for intervenor State of Michigan.

Before: EDGERTON, BAZELON, and BURGER, Circuit Judges.

EDGERTON, Circuit Judge:

Petitioners seek review of the Atomic Energy Commission's Order of May 26, 1959 which continued in effect, with amendments, a "provisional" construction permit issued August 4, 1956, for a nuclear power reactor. Section 104b of the Atomic Energy Act of 1954 authorizes the Commission to issue licenses for "utilization and production facilities involved in the conduct of research and development activities leading to the demonstration of the practical value of such facilities for industrial or commercial purposes. In issuing licenses under this subsection, the Commission shall impose the minimum amount of such regulations and terms of license as will permit the Commission to fulfill its obligations under this Act to promote the common defense and security and to protect the health and safety of the public . . . ." 68 Stat. 937, 42 U.S.C. §2134(b).

The holder of the construction permit, intervenor here, is the Power Reactor Development Company (PRDC), a Michigan membership corporation organized "to study, develop, design, fabricate, construct and operate one or more experimental nuclear power reactors . . . to the

end that there may be an early demonstration of the practical and economical use of nuclear energy for the generation of electrical energy . . ." Of PRDC's 21 members, 14 are public utilities and 7 are equipment manufacturers.

The reactor will be the largest, but not the first, "fast breeder" reactor in the United States. The site is at Lagoona Beach, Monroe County, Michigan, on the shore of Lake Erie, 30 miles southwest of Detroit.

[fol. 6933] We cannot review the Commission's order at petitioners' request unless (1) it is a "final order" and (2) petitioners are "aggrieved" by it. Atomic Energy Act of 1954, § 189, 42 U.S.C. § 2239(b); 5 U.S.C. §§ 1032, 1034. Although the Commission's action of May 26, 1959 was entitled "Commission's Opinion, Final Decision and Order," the Commission and PRDC now contend that the order was not final. They also contend that it did not aggrieve the petitioners. In our opinion it was what it purported to be, a final order, and petitioners are "aggrieved" by it. Because it threatens them with economic injury, they "had the requisite standing to appeal and to raise . . . any relevant question of law in respect of the order . . ." *Federal Communications Commission v. Sanders Brothers Radio Station*, 309 U.S. 470, 477.

Petitioners are national or international labor unions which intervened, with some of their members, in the proceedings before the Commission, on the basis that "granting the conditional construction permit herein (1) is a violation of the provisions of the Atomic Energy Act of 1954, and the regulations pursuant thereto adopted by the Commission . . . and (2) will result in the construction of a reactor which, under present technological conditions, is inherently unsafe, and which will thereby create a hazard which will place the individual Intervenors, the members of the UAW and their families, and the UAW in danger of an explosion or other incident" damaging to the individuals and their homes, real estate values, and employment; that the value of collective bargaining contracts "will be seriously impaired if the PRDC reactor is built in this area without reasonable assurances of safety"; and that there are "reasonable grounds for

belief that a license to operate said facility when it is completed, with an expenditure of $45,000,000 will be issued without proper consideration of and regard for the health and safety of the public."

[fol. 6934] In their reply brief in this court, petitioners contend that "The fear of a possible atomic catastrophe, in itself, before any operation would begin, would; among other things have the effect of depressing values of property owned by the Petitioners, and would cause plants in which they work under collective bargaining agreements to move and thereby cause a loss of employment." Their reply brief asserts that "the uncontroverted allegations of their petition for intervention before the Commission set forth the economic injury they would suffer merely from the construction of the reactor itself." But we find no such allegations in their petition for intervention before the Commission. The theory of that petition was that construction would cause operation, and operation would cause injury, not that construction without operation would cause injury.[1] Judicial review is limited to the record before the Commission. 5 U.S.C. § 1037(a).

As the Commission says in its order, "a construction permit is a step toward a license rather than the equivalent thereof. . . . This permit is provisional to the extent that a license authorizing operation of the facility will not be issued by the Commission unless PRDC has submitted to the Commission (by proposed amendment to the Application) the complete, final Hazards Summary Report (portions of which may be submitted and evaluated from time to time), and the Commission has found that the final design provides reasonable assurance that the health and safety of the public will not be endangered by operation of the facility in accordance with the specified procedures. It is further provisional to the extent that the Commission reserves jurisdiction, at any time prior to issuance of an operating license, upon notice to the parties herein, to reopen this proceeding for the purpose of receiving additional evidence, and to make such

[1] It is undisputed that construction without operation will cause no *physical* injury or danger not involved in the erection of any large building.

[fol. 6935] determinations and take such action with respect to the continuance, vacation, or modification of this permit as the entire record warrants." But the order also says: "There is reasonable assurance that theoretical and experimental programs under way will develop sufficient data to justify the issuance of an operating license, and that the results of these programs will be available prior to the time it is necessary for the Commission to rule on the operating aspect of the PRDC license Application." PRDC says "it must be taken as settled . . . that the further technical information needed to complete the PRDC application for license will be supplied." Although this positive prediction overstates the matter, it is plainly probable, in a high degree, that if the construction permit stands PRDC will get an operating license and will operate. We think petitioners are therefore aggrieved by the issuance of the permit.

Safety findings required by the Atomic Energy Act

Petitioners contend that "The Act and the regulations of the Commission . . . require, as conditions precedent to the issuance of every construction permit for an atomic energy power reactor, that *as of the time the construction permit is issued* the Commission find that (1) it has reasonable assurance that the reactor may be constructed *and operated* at the proposed site without undue risk to the health and safety of the public . . . ."

It is undisputed that the Commission must make such a finding when it authorizes operation. The question is whether it must make such a finding when it authorizes construction. In our opinion it must.

Section 182 of the Atomic Energy Act of 1954, which is headed "License applications", provides in paragraph (a): " . . . In connection with applications for licenses to operate production or utilization facilities, the applicant shall state such technical specifications, including . . . the place of the use . . . and such other information as [fol. 6936] the Commission may, by rule or regulation, deem necessary in order to enable it to find that the utilization or production of special nuclear material . . . will provide adequate protection to the health and safety of the pub-

lic. Such technical specifications shall be a part of any license issued. . . ." 42 U.S.C. § 2232(a).

It seems to be unquestioned that the phrase used in § 182, "adequate protection to the health and safety of the public", and the Commission's phrase, "without undue risk to the health and safety of the public", are substantially equivalent.

Section 185 of the Act, which is headed "Construction permits", provides: "All applicants for licenses to construct or modify production or utilization facilities shall, if the application is otherwise acceptable to the Commission, be initially granted a construction permit. . . . Upon the completion of the construction or modification of the facility, upon the filing of any additional information needed to bring the original application up to date; and upon finding that the facility authorized has been constructed and will operate in conformity with the application as amended and in conformity with the provisions of this chapter and of the rules and regulations of the Commission, and in the absence of any good cause being shown to the Commission why the granting of a license would not be in accordance with the provisions of this Act, the Commission shall thereupon issue a license to the applicant. For all other purposes of this Act a construction permit is deemed to be a 'license'." 42 U.S.C. § 2235.

While the bill was pending, Senator Humphrey proposed, and withdrew, an amendment which would have added after the word "license", at the end of § 185: " , and no construction permit shall be issued by the Commission until after the completion of the procedures established by Section 182 for the consideration of applications for licenses under this act." (100 Cong. Rec. 11566 [fol. 6937] (1954); Legislative History of the Atomic Energy Act of 1954, Vol. III, p. 3759; Vol. I, p. 733) He said: "*The purpose of the amendment* when it was prepared *was to make sure that the construction of a facility was not permitted prior to the authorization of a license*, because had that been done what it would have amounted to would be getting an investment of a substantial amount of capital, which surely would have been prejudicial in terms

of the Commission issuing the license. In other words, if the Commission had granted the construction permit for some form of nuclear reactor, and then the question of a license was not fully resolved, surely there would have been considerable pressure, and justifiably so, for the Commission to have authorized the license once it had authorized the permit for construction.

"The chairman of the committee tells me he has modified certain sections by the committee amendments to the bill, of which at that time I was not aware. *The chairman indicates to me that under the terms of the bill, as amended, the construction permit is equivalent to a license.* In other words, as I understand, under the bill a construction permit cannot be interpreted in any other way than being equal to or a part of the licensing procedure. Is that correct?" Senator Hickenlooper, the manager of the bill, replied: "The Senator is correct. The staff has worked on this matter. . . . A license and a construction permit are equivalent. . . . Therefore, we believe, and we assure the Senator, that *the amendment is not essential to the problem which he is attempting to reach.*" After some discussion of other sections of the bill, this colloquy occurred: "Mr. HUMPHREY. In other words, the revised sections on judicial review and on hearings and *the revised section 182 on license application all apply directly to construction permit?* Mr. HICKENLOOPER. *Yes.* Mr. HUMPHREY. *With that statement, Mr. President, I withdraw my amendment. The only purpose of the amendment was to clarify that* [fol. 6938] *section.* I am grateful to the chairman for having done it before the amendment was considered." (Emphasis added.) (100 Cong. Rec. 11566; Legislative History, Vol. III, p. 3759.)[2]

If, as this indicates, § 182 applies "directly to" construction permits, when the Commission issues a construction permit it must "find that the utilization or production of special nuclear material . . . will provide adequate

[2] The Commission apparently interprets this colloquy as concerning only the "procedural safeguards" of notice, hearings, and appeal. We cannot so understand it and cannot suppose the Senate so understood it.

protection to the health and safety of the public"; or, in the Commission's phrase, that the facility can be "operated at the location proposed without undue risk to the health and safety of the public."

The Joint Committee on Atomic Energy said in its report on the bill: "Section 185 permits the Commission to issue construction permits to applicants for a production or utilization facility, describes the terms of the construction permit, and requires the issuance of a license if the construction is carried out in accordance with the terms of the construction permit." (S.Rep.No. 1699, 83d Cong., 2d Sess., 28 (1954); Legislative History, Vol. I, p. 776.) It seems certain that if the Act did not require, as a condition to the issuance of a construction permit, a finding that the proposed facility can be operated without undue risk to the health and safety of the public, the Act would not require the issuance of a license when the permitted construction is carried out.

At the very least it is doubtful whether the Commission's construction of the Atomic Energy Act is correct. The possibilities of harm are so enormous that any doubt as to what findings the Act requires, and any doubt as to whether the Commission made such findings, should be resolved on the side of safety.

[fol. 6939] *The Commission's Safety Findings*

In our opinion the Commission's findings regarding safety of operation are not sufficient.

An Initial Decision dated December 10, 1958, contains this unqualified finding: "22. The Commission finds reasonable assurance in the record that a utilization facility of the general type proposed in the PRDC application and amendments thereto can be constructed, and will be able to be operated at the location proposed without undue risk to the health and safety of the public." But in the Opinion and Final Decision which accompanied its order of May 26, 1959, by interpolating the phrase we emphasize, the Commission qualified the finding: "22. The Commission finds reasonable assurance in the record, *for the purposes of this provisional construction permit*, that

a utilization facility of the general type proposed in the PRDC Application and amendments thereto can be constructed and operated at the location without undue risk to the health and safety of the public." (Emphasis added.) This is not a finding that a facility can be operated there without undue risk. It is a finding that there is sufficient likelihood that a facility can be operated there without undue risk so that, in the Commission's opinion, it is appropriate to issue a "provisional" construction permit. In our opinion such a finding does not meet the requirements of the Act.

The Commission made other statements which confirm the impression that it no longer found, as it had found in December, reasonable assurance that a facility can be operated at the location without undue risk. The Commission said: "The degree of 'reasonable assurance' with respect to safety that satisfies us in this case for purposes of the *provisional* construction permit would not be the same as we would require in considering the issuance of the *operating* license. . . . It has not been positively established that a fast breeder reactor of the general type and power level proposed by Applicant can be *oper-* [fol. 6940] *ated* without a credible possibility of releasing significant quantities of fission products to the environment . . . ." (Emphasis in original.) And again: "*For the purposes of a provisional construction permit*, there is reasonable assurance that a reactor of the general type described in the Application can be so designed that no credible accident in the course of its operation is likely to result in the release of significant quantities of fission products into the atmosphere." (Emphasis added.)

The Commission expressed confidence that future scientific developments would enable it, in the future, to find that the reactor could be operated without undue risk. It said: "There is reasonable assurance that theoretical and experimental programs under way will develop sufficient data to justify the issuance of an operating license, and that the results of these programs will be available prior to the time it is necessary for the Commission to rule on the operating aspect of the PRDC license Application." "There is reasonable assurance that theoretical

and experimental investigations which have been undertaken, together with operating experience on one or more of the EBR-I, EBR-II and Dounreay reactors, will establish definitively, prior to the scheduled completion date of the PRDC reactor, whether or not the reactor proposed by Applicant can be so operated"; *i.e.,* whether it can be "operated without a credible possibility of releasing significant quantities of fission products to the environment." Again, "there is reasonable assurance that *evidence will establish* that the reactor proposed by Applicant can be so operated." (Emphasis added.) This clearly implies that *evidence does not now establish* that the reactor can be so operated. The Commission's predictions regarding the future course of scientific development do not in our opinion satisfy the requu.rement of the Act.

The Commission said: "It is in the nature of reactor design, although certainly not unique to it, that many [fol. 6941] features remain to be designed and demonstrated after construction is begun—and indeed some features redesigned and replaced after operation is under way. . . . By proceeding with construction and further research and development simultaneously, rather than awaiting complete research and development results Applicant will save several years in the time required to place in operation its demonstration power reactor." As a matter of policy, there is force in these considerations. But Congress seems to have been more impressed by the opposite policy considerations to which Senator Humphrey, in his colloquy with Senator Hickenlooper, called the attention of the Senate. The economy cannot afford to invest enormous sums in the construction of an atomic reactor that will not be operated. If enormous sums are invested without assurance that the reactor can be operated with reasonable safety, pressure to permit operation without adequate assurance will be great and may be irresistible. PRDC's estimate of the cost of construction, preconstruction research and development, and administrative expenses during construction and test operation was $44,-020,000. The Commission found there would probably be "a cost over-run".

In contrast with the Commission's repeated expressions of uncertainty, it used other expressions which might seem to indicate a positive opinion regarding safety of operation. The Opinion and Final Decision, before adverting to the issue of safety and other issues, said broadly: "we amplify and affirm our Opinion and Initial Decision dated December 10, 1958." The Commission also said: "The principal factual issue in this proceeding is whether there is information sufficient to provide a reasonable assurance that a utilization facility of the general type proposed in the PRDC application can be constructed and operated at the location proposed therein without undue risk to the health and safety of the public. Subsidiary to this issue is whether there is reasonable [fol. 6942] assurance that technical information omitted from, and required to complete, the application will be supplied before issuance of an operating license. A careful evaluation of the entire record in this proceeding can only lead to an affirmative answer to all of these questions." And again: "It is enough for the purposes of the present proceeding (that is, for the issuance of a provisional construction permit), and for the satisfaction of the requirements of the statute and the regulations, that there be reasonable assurance that the reactor can be constructed and operated without undue risk to the health and safety of the public. We conclude that the present state of knowledge as described in the record gives, and the accident possibilities presented on the record do not negate, that assurance."

It results that the Commission's findings regarding safety of operation are ambiguous. In view of the nature, size, and location of the project, we think the findings should be uncommonly free from ambiguity. The Commission should "make the basis of its action reasonably clear. We cannot find that it did so here." *Radio Station KFH Co. v. Federal Communications Commission*, 101 U.S.App.D.C. 164, 166, 247 F. 2d. 570, 572. "'We must know what a decision means before the duty becomes ours to say whether it is right or wrong.'" *Secretary of Agriculture v. United States*, 347 U.S. 645, 654.

Pacific Far East Line, Inc. v. *Federal Maritime Board*, —— U.S.App.D.C. ——, ——, 275 F. 2d 184, 187.

We think the Commission's safety findings are deficient in an additional respect.

In 1957 the Commission made to the Joint Committee on Atomic Energy "a report of a study of the possible consequences in terms of injury to persons and damage to property, if certain hypothetical major accidents should occur in a typical large nuclear power reactor." All the experts agreed "that the chances that major accidents might occur are exceedingly small." But "Under adverse [fol. 6943] combinations of the conditions considered, it was estimated that people could be killed at distances up to fifteen miles, and injured at distances of about forty-five miles. Land contamination could extend for greater distances." Undisputed testimony before the Commission shows that there is a "possibility of a major disaster, even though it has a low probability".

As the Commission said, "the question of safety obviously cannot be considered without regard to proposed location." The Commission found: "The site is bordered on one side by water and provides an exclusion area on the land side with a minimum radius of 2900 feet. The population distribution for given distances from the site is as follows: 1 mile, population 175; 2, 600; 5, 1,800; 10, 31,300; 20, 187,100; 30, 2,001,700. During the summer months the population within five miles would be increased due to vacationing transients and to the fact that beaches two to five miles southwest of the site may be crowded with thousands of people."

We think it clear from the Congressional concern for safety that Congress intended no reactor should, without compelling reasons, be located where it will expose so large a population to the possibility of a nuclear disaster. It does not appear that the Commission found compelling reasons or saw that such reasons were necessary. It said: "The evidence of record with respect to site gives reasonable assurance that the site is satisfactory from structural and underground water flow standpoints. The meteorology of the site is complex, but no reason appears in the record for it to be disqualifying. The site makes

possible extensive safeguards against the inadvertent release of liquid contaminants. . . . Studies of weather, hydrology, geology, and similar problems have yielded considerable information and are still in progress. Although the data of these types are not yet complete or conclusive, the record gives reasonable assurance that safe operation of the reactor will be as likely in that [fol. 6944] location as in any other location."[3] We think this finding clearly insufficient. We need not consider whether even the most compelling reasons for preferring this location could support a finding that the reactor could be operated at this location without "undue" risk, or with "adequate" protection, to the health and safety of the public.

Because we think the safety findings insufficient, we must set aside the Commission's grant of a construction permit and remand the case for such further proceedings consistent with this opinion as the Commission may determine. We need not consider other points raised by the petitioners.

BURGER, Circuit Judge, dissenting:

I dissent because I think there is no occasion at this time for the court to reach the issue of the ultimate safety of the plant's operations. The Commission has issued only a provisional permit to build a plant, not to operate it. The plant cannot go into operation until and unless the intervenor PRDC meets the safety provisions of the Act.

The sole basis of challenge to the provisional construction permit is that the *future possibility* that an operating permit will be unlawfully and improperly issued by the

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[3] The Commission continued: "We anticipate that knowledge to be acquired will fortify that assurance. . . . It is possible that there may be presently unknown effects in large fast reactor systems. A prototype of the proposed reactor at a remote location has been urged as affording greater assurance against the possibility of such unknown effects than does the presently planned experimental and theoretical programs. (sic) The Commission finds that the necessity, however, for constructing such a prototype has not been shown. If the program of meltdown investigation should prove inconclusive, it will be necessary to reconsider the question of need for a prototype."

[fol. 6945] Commission creates a "present," "immediate" and "unavoidable threat" of injury. I do not think we have any occasion to consider what is not now before us. The Commission expressly deferred action on that issue. This does not appear to me a "final order" which gives us jurisdiction to pass on the ultimate issue of safety: nor does it empower us to tell the Commission that it must pass on the ultimate safety of the operation before the plant is constructed. Orders are not final as to a person "unless and until they impose an obligation, deny a right or fix some legal relationship as a consummation of the administrative process." *Chicago & Southern Air Lines, Inc.* v. *Waterman Steamship Corp.*, 333 U.S. 103, 113 (1948).

In an area involving as much scientific uncertainty as development of nuclear energy for peaceful purposes, the Commission must be permitted to proceed step by step, *i.e.*, make its preliminary finding of probable safety when the construction permit issues and reserve final approval of operations until a later date.

I respectfully suggest that my colleagues are undertaking to assume responsibilities which Congress vested in the Commission. This is illustrated in the majority's statement.

> "No reactor should, without compelling reasons, be located where it will expose so large a population to the possibility of a nuclear disaster."

On what evidence does the majority make a finding of "nuclear disaster" directly opposed to the finding which the Atomic Energy Commission made? The majority is, in effect, telling the Atomic Energy Commission that it has made an *unwise* decision on the location of the plant.

The majority also goes beyond the established limits of judicial review when it states:

> "The economy cannot afford to invest enormous sums in the construction of an atomic reactor that will not be operated. If enormous sums are invested without [fol. 6946] assurance that the reactor can be operated with reasonable safety, pressure to permit operation

without adequate assurance will be great and may be irresistible."

From an erroneous premise drawn out of thin air, the majority proceeds to draw an unwarranted conclusion. On what evidence can we as judges say our "economy cannot afford," or even that these appellees cannot afford, this large investment for peaceful uses of nuclear energy? I suggest our entire history is to the contrary. We invested not mere millions but *billions* in the original development of nuclear fission on a totally unproven theory of physics. It was an act of faith in the views of scientists. Surely it cannot be seriously suggested that these giants of American industry which formed PRDC are not well able—and willing—to risk the loss of millions in experiments and research. Forty or fifty million dollars to the sponsors of PRDC is a small investment to risk for the world's first known experiment of this kind into peaceful uses of nuclear energy.

If we were dealing with a radio or TV license or some other purely commercial enterprise in a developed and mature industry I would agree that there is a risk that large investment in machines might conceivably exert a subtle influence on the ultimate grant of an operations permit. Cf. *Community Broadcasting Co. v. Federal Communications Commission*, Nos. 15313, 15314 (D.C.Cir., Feb. 8, 1960). But I cannot join in the suggestion that members of the Atomic Energy Commission who have assumed obligations under oaths as binding as ours would permit an operation dangerous to the public because 40 or 50 million dollars is invested in brick, mortar and steel by men who knew from the outset they were engaged in a scientific gamble. And if any administrative agency should so abdicate its responsibilities in a matter as grave as this—which I cannot believe is likely—the courts are always in a position to exercise a final and stringent scrutiny on the issue of public safety.

[fol. 6947] Development in an area like this must, of necessity, proceed step by step. The Commission has found that the issuance of the construction permit on a provisional and restricted basis "does not in any manner adversely

affect the health and safety of the public or that of the [petitioners]." The appellants do not attack this finding.

At this stage how can anyone know what the result will be? This court considered a challenge not unlike that of appellants' challenge in *Associated-Banning Co. v. United States*, 247 F.2d 557, 561 (1957): "We cannot assume that the Board will not conduct its hearing within the intendment of the Act, so far as it may apply. It may well be, for all we are shown, that the Board's ultimate action will completely dispel every prospective fear voiced by the protest and the complaint. It is clear that the Board has not as yet entered an order 'final' as to these petitioners."

The essence of the majority action is found in its acceptance of the idea that once the Commission has permitted PRDC to invest its millions in the plant they are "bound" or "likely" to relax their notion of what is safe or dangerous in order to bail out the investors.

I emphasize that I cannot for a moment believe the sponsors of PRDC are so naive that they would think their investment of these millions is not speculative just as is most research. Nor can I believe they think that any amount of invested capital will persuade the Atomic Energy Commission to make a finding of safety which is not supported by substantial scientific evidence. It is entirely possible that PRDC might find itself the owner of a 50 million dollar scientific "white elephant" if, after completion of construction, it cannot satisfy the safety standards of the statute. Should that be the case it will be simply one of the unproductive steps in what promises to be a program to open to mankind sources of power undreamed of only a few years ago.

[fol. 6948] [File endorsement omitted]

IN UNITED STATES COURT OF APPEALS

FOR THE DISTRICT OF COLUMBIA CIRCUIT

No. 15,271—September Term, 1959

INTERNATIONAL UNION OF ELECTRICAL, RADIO AND MACHINE WORKERS, AFL-CIO; UNITED AUTOMOBILE, AIRCRAFT AND AGRICULTURAL IMPLEMENT WORKERS OF AMERICA; and UNITED PAPERMAKERS AND PAPERWORKERS, Petitioners,

v.

UNITED STATES OF AMERICA and ATOMIC ENERGY COMMISSION, Respondents,

POWER REACTOR DEVELOPMENT COMPANY, STATE OF MICHIGAN, Intervenors.

On Petition to Review an Order of the Atomic Energy Commission.

Before: Edgerton, Bazelon, and Burger, Circuit Judges.

JUDGMENT—June 10, 1960

This case came on to be heard on the record from the Atomic Energy Commission, and was argued by counsel.

On Consideration Whereof, It is ordered and adjudged by this court that the order of the Atomic Energy Commission on review in this case is set aside, and that this case is remanded to the said Atomic Energy Commission for such further proceedings consistent with the opinion of this court as the Commission may determine.

Per Circuit Judge Edgerton.

Dated: June 10, 1960.

Separate dissenting opinion by Circuit Judge Burger.

[fol. 6949] [File endorsement omitted]

IN UNITED STATES COURT OF APPEALS
FOR THE DISTRICT OF COLUMBIA CIRCUIT

[Title omitted]

Before: Prettyman, Chief Judge, Edgerton, Wilbur K. Miller, Bazelon, Fahy, Washington, Danaher, Bastian and Burger, Circuit Judges, in Chambers.

ORDER DENYING PETITION OF POWER REACTOR DEVELOPMENT COMPANY FOR A REHEARING IN BANC—July 25, 1960

Upon consideration of the petition of intervenor Power Reactor Development Company for a rehearing in banc, it is

Ordered by the court that the aforesaid petition is denied.

Per Curiam.

Dated: July 25, 1960.

Circuit Judges Miller and Bastian would grant the petition for rehearing in banc.

Circuit Judges Washington and Burger did not participate in this order.

[fol. 6950] [File endorsement omitted]

IN UNITED STATES COURT OF APPEALS
FOR THE DISTRICT OF COLUMBIA CIRCUIT

[Title omitted]

Before: Prettyman, Chief Judge, Edgerton, Wilbur K. Miller, Bazelon, Fahy, Washington, Danaher, Bastian and Burger, Circuit Judges, in Chambers.

ORDER DENYING PETITION OF RESPONDENTS FOR A REHEARING IN BANC—July 25, 1960

Upon consideration of respondents' petition for a rehearing in banc, it is

Ordered by the court that the petition for rehearing in banc is denied.

Per Curiam.

Dated: July 25, 1960.

Circuit Judges Miller and Bastian would grant the petition for rehearing in banc.

Circuit Judges Washington and Burger did not participate in this order.

[fol. 6951] [File endorsement omitted]

IN UNITED STATES COURT OF APPEALS

FOR THE DISTRICT OF COLUMBIA CIRCUIT

[Title omitted]

Before: Fahy, Acting Chief Judge, in Chambers.

ORDER STAYING TRANSMISSION OF CERTIFIED COPIES OF OPINION AND JUDGMENT—August 8, 1960

Upon consideration of intervenor's motion for stay of transmission of certified copies of the opinion and judgment of this court, and it appearing that no objections have been filed, it is

Ordered that transmission of the certified copies of the opinion and judgment of this court is stayed to and including August 15, 1960.

Dated: August 8, 1960.

[fol. 6952] Clerk's Certificate to Foregoing Transcript (omitted in printing).

[fol. 6953]

SUPREME COURT OF THE UNITED STATES

No. 315—October Term, 1960

POWER REACTOR DEVELOPMENT COMPANY, Petitioner,

vs.

INTERNATIONAL UNION OF ELECTRICAL, RADIO AND MACHINE WORKERS, AFL-CIO, et al.

ORDER ALLOWING CERTIORARI—November 14, 1960

The petition herein for a writ of certiorari to the United States Court of Appeals for the District of Columbia Circuit is granted, limited to Questions 1 and 2 presented by the petition for writ of certiorari. The case is consolidated with No. 454 and a total of two hours is allotted for oral argument.

And it is further ordered that the duly certified copy of the transcript of the proceedings below which accompanied the petition shall be treated as though filed in response to such writ.

[fol. 6954]

SUPREME COURT OF THE UNITED STATES

No. 454—October Term, 1960

UNITED STATES et al., Petitioners,

vs.

INTERNATIONAL UNION OF ELECTRICAL, RADIO AND MACHINE WORKERS, AFL-CIO, et al.

ORDER ALLOWING CERTIORARI—November 14, 1960

The petition herein for a writ of certiorari to the United States Court of Appeals for the District of Columbia Circuit is granted, limited to Questions 1 and 2 presented by the petition for writ of certiorari. The case is consolidated with No. 315 and a total of two hours is allotted for oral argument.

And it is further ordered that the duly certified copy of the transcript of the proceedings below which accompanied the petition shall be treated as though filed in response to such writ.